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JUL 16 2012

10 CFR 50.46

LR-N12-0220

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

Salem Nuclear Generating Station Units 1 and 2
Facility Operating License Nos. DPR-70 and 75
NRC Docket Nos. 50-272 and 50-311

Subject: **SALEM LOSS OF COOLANT ACCIDENT PEAK CLADDING TEMPERATURE MARGIN TRACKING-ANNUAL REPORT**

REFERENCE: 1) Westinghouse Letter LTR-LIS-12-108, "Salem Units 1 and 2 10CFR50.46 Annual Notification and Reporting for 2011," February 24, 2012.

2) PSEG Letter LR-N11-0211, "Salem Nuclear Generating Station Units 1 and 2 Facility Operating License Nos. DPR-70 and 75 NRC Docket Nos. 50-272 and 50-311, Salem Loss of Coolant Accident Peak Cladding Temperature Margin Tracking -Annual Report," July 18, 2011.

In accordance with 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," paragraph (a)(3)(ii), PSEG Nuclear is required to submit an annual report of the Emergency Core Cooling System (ECCS) Evaluation Model changes and errors for Salem Units 1 and 2.

For this reporting period of July 2011 to June 2012, there have been various issues identified via Reference 1; however, no changes to the PCT rack-ups from 2011 are required. The PCT rack-ups are being sent for completeness only. The previous Peak Cladding Temperature (PCT) report PSEG Nuclear filed with the NRC for Salem was dated July 18, 2011 (Reference 2).

Attachment 1, "Peak Cladding Temperature Rack-Up Sheets," provides updated information regarding the PCT for the limiting small break and large break loss-of-coolant accident (LOCA) evaluations for Salem Units 1 and 2.

Attachment 2, "Assessment Notes," contains a detailed description for each of these previous changes or errors reported.

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On December 13, 2011, the NRC issued Information Notice 2011-21: "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting From Nuclear Fuel Thermal Conductivity Degradation." This Information Notice was reviewed and found to not be directly applicable to the Salem plants since we utilize non-Best Estimate LBLOCA methods (Appendix K, BASH LBLOCA methodology).

Subsequent discussions were held with Westinghouse, the NRC, and the PWR industry through the PWR Owners Group. Westinghouse has informed PSEG Nuclear that all of their LBLOCA methods, including the Appendix K, BASH method used for Salem, are adversely impacted by Thermal Conductivity Degradation (TCD). The PWR Owners Group launched a generic project (PA-ASC-1073) to determine LBLOCA PCT impact due to TCD for all plants using any Westinghouse LBLOCA method. This project is on track to provide Salem with the LBLOCA PCT impact in mid-August 2012. It is expected that Salem has sufficient PCT margin to accommodate the TCD impact.

There are no commitments made in this letter. If you have any questions regarding this letter, please contact Chris Dahms at (856) 339-5456.

Sincerely,

Carl J. Fricker
Site Vice President – Salem

cc: Mr. W. Dean, Administrator, Region I, NRC
Mr. J. Hughey, Project Manager, NRC
NRC Senior Resident Inspector, Salem
Mr. P. Mulligan, Manager IV, NJBNE
Mr. L. Marabella, Corporate Commitment Tracking Coordinator
Mr. T. Cachaza, Salem Commitment Tracking Coordinator

Attachments (2)

Attachment 1
LR-N12-0220
Peak Cladding Temperature Rack-Up Sheets

SALEM UNITS 1 AND 2

Docket Nos. 50-272 and 50-311

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

Report of the Emergency Core Cooling System Evaluation Model Changes and Errors Assessments

Attachment 1
LR-N12-0220
Peak Cladding Temperature Rack-Up Sheets

PLANT NAME: Salem Unit 1
ECCS EVALUATION MODEL: Small Break Loss of Coolant Accident (SBLOCA)
REPORT REVISION DATE: 6/19/12
CURRENT OPERATING CYCLE: 22

ANALYSIS OF RECORD (AOR)

Evaluation Model: NOTRUMP
Calculation: Westinghouse PSE-93-568, March 1993
Fuel: RFA 17 x 17
Limiting Fuel Type: RFA 17x17
Heat Flux Hot Channel Factor (F_Q) = 2.4
Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$) = 1.65
Steam Generator Tube Plugging = 10%
Limiting Break Size: 2 inches
Break Location: Cold Leg
Limiting Single Failure: loss of one train of ECCS flow
Reference Peak Cladding Temperature (PCT) PCT = 1580°F

MARGIN ALLOCATION

A. PRIOR LOSS OF COOLANT ACCIDENT (LOCA) MODEL ASSESSMENTS

10 CFR 50.46 report dated October 29, 1993 (See Note 1)	ΔPCT = -13°F
10 CFR 50.46 report dated July 27, 1994 (See Note 2)	ΔPCT = -16°F
10 CFR 50.46 report dated December 8, 1994 (See Note 3)	ΔPCT = +109°F
10 CFR 50.46 report dated January 18, 1995 (See Note 4)	ΔPCT = 0°F
10 CFR 50.46 report dated December 7, 1995 (See Note 5)	ΔPCT = 0°F
10 CFR 50.46 report dated August 2, 1996 (See Note 6)	ΔPCT = -8°F
10 CFR 50.46 report dated July 11, 1997 (See Note 7)	ΔPCT = 0°F
10 CFR 50.46 report dated June 10, 1998 (See Note 8)	ΔPCT = 0°F
10 CFR 50.46 report dated April 27, 1999 (See Note 9)	ΔPCT = 0°F
10 CFR 50.46 report dated October 18, 1999 (See Note 10)	ΔPCT = +10°F
10 CFR 50.46 report dated September 21, 2000 (See Note 11)	ΔPCT = +27°F
10 CFR 50.46 report dated August 27, 2001 (See Note 12)	ΔPCT = 0°F
10 CFR 50.46 report dated August 27, 2002 (See Note 13)	ΔPCT = 0°F
10 CFR 50.46 report dated August 08, 2003 (See Note 14)	ΔPCT = 0°F
10 CFR 50.46 report dated July 29, 2004 (See Note 15)	ΔPCT = +40°F
10 CFR 50.46 report dated July 28, 2005 (See Note 16)	ΔPCT = 0°F
10CFR 50.46 report dated July 28, 2006 (See Note17)	ΔPCT = 0°F
10CFR 50.46 report dated July 25, 2007 (See Note 18)	ΔPCT = 0°F
10CFR 50.46 report dated July 22, 2008 (See Note 19)	ΔPCT = 0°F
10CFR 50.46 report dated July 20, 2009 (See Note 20)	ΔPCT = 0°F
10CFR 50.46 report dated July 20, 2010 (See Note 21)	ΔPCT = 0°F
10CFR 50.46 report dated July 18, 2011 (See Note 22)	ΔPCT = 0°F

NET PCT **PCT = 1729°F**

B. CURRENT LOCA MODEL ASSESSMENTS

Radiation Heat Transfer Logic (See Note 23)	ΔPCT = 0°F
Maximum Fuel Rod Time Step Logic (See Note 24)	ΔPCT = 0°F
General Code Maintenance (NOTRUMP) (See Note 25)	ΔPCT = 0°F
Total PCT change from current assessments	Σ ΔPCT = 0°F
Cumulative PCT change from current assessments	Σ |ΔPCT| = 0°F

NET PCT **PCT = 1729°F**

Attachment 1
LR-N12-0220
Peak Cladding Temperature Rack-Up Sheets

PLANT NAME: Salem Unit 1
ECCS EVALUATION MODEL: Large Break Loss of Coolant Accident (LBLOCA)
REPORT REVISION DATE: 6/19/12
CURRENT OPERATING CYCLE: 22

ANALYSIS OF RECORD (AOR)

Evaluation Model: BASH
Calculation: Westinghouse 93-PSE-G-0080, September 1993
Fuel: RFA 17 x 17
Limiting Fuel Type: RFA 17x17
Heat Flux Hot Channel Factor (F_Q) = 2.4
Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$) = 1.65
Steam Generator Tube Plugging = 10%
Limiting Break Size: Cd = 0.4
Break Location: Cold leg
Limiting Single Failure: Loss of one train of ECCS flow

Reference Peak Cladding Temperature (PCT) PCT = 1978°F

MARGIN ALLOCATION

A. PRIOR LOSS OF COOLANT ACCIDENT (LOCA) MODEL ASSESSMENTS

10 CFR 50.46 report dated January 18, 1995 (See Note 4)	ΔPCT = +36°F
10 CFR 50.46 report dated December 7, 1995 (See Note 5)	ΔPCT = 0°F
10 CFR 50.46 report dated August 2, 1996 (See Note 6)	ΔPCT = 0°F
10 CFR 50.46 report dated July 11, 1997 (See Note 7)	ΔPCT = +15°F
10 CFR 50.46 report dated June 10, 1998 (See Note 8)	ΔPCT = 0°F
10 CFR 50.46 report dated April 27, 1999 (See Note 9)	ΔPCT = 0°F
10 CFR 50.46 report dated October 18, 1999 (See Note 10)	ΔPCT = +12°F
10 CFR 50.46 report dated September 21, 2000 (See Note 11)	ΔPCT = +9°F
10 CFR 50.46 report dated August 27, 2001 (See Note 12)	ΔPCT = +6°F
10 CFR 50.46 report dated August 27, 2002 (See Note 13)	ΔPCT = +20°F
10 CFR 50.46 report dated August 08, 2003 (See Note 14)	ΔPCT = +7°F
10 CFR 50.46 report dated July 29, 2004 (See Note 15)	ΔPCT = +5°F
10 CFR 50.46 report dated July 28, 2005 (See Note 16)	ΔPCT = 0 °F
10 CFR 50.46 report dated July 28, 2006 (See Note 17)	ΔPCT = -50 °F
10 CFR50.46 report dated July 25, 2007 (See Note 18)	ΔPCT = +4°F
10CFR 50.46 report dated July 22, 2008 (See Note 19)	ΔPCT = 0°F
10CFR 50.46 report dated July 20, 2009 (See Note 20)	ΔPCT = 0°F
10CFR 50.46 report dated July 20, 2010 (See Note 21)	ΔPCT = 0°F
10CFR 50.46 report dated July 18, 2011 (See Note 22)	ΔPCT = 0°F

NET PCT **PCT = 2042°F**

B. CURRENT LOCA MODEL ASSESSMENTS

General Code Maintenance (BASH) (See Note 25)	ΔPCT = 0°F
Total PCT change from current assessments	$\Sigma\ \Delta$PCT = 0°F
Cumulative PCT change from current assessments	$\Sigma\ \lvert\Delta\text{PCT}\rvert$ = 0°F

NET PCT **PCT = 2042°F**

Peak Cladding Temperature Rack-Up Sheets

PLANT NAME: Salem Unit 2
ECCS EVALUATION MODEL: Small Break Loss of Coolant Accident (SBLOCA)
REPORT REVISION DATE: 6/19/12
CURRENT OPERATING CYCLE: 19

ANALYSIS OF RECORD (AOR)

Evaluation Model: NOTRUMP
Calculation: Westinghouse (PSE-04-131), December 2004
Fuel: RFA 17 x 17
Limiting Fuel Type: RFA 17x17
Heat Flux Hot Channel Factor (F_Q) = 2.5
Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$) = 1.65
Steam Generator Tube Plugging = 10%
Limiting Break Size: 3 inches
Break Location: Cold Leg
Single Failure: loss of one train ECCS flow

Reference Peak Cladding Temperature (PCT) PCT = 987°F

MARGIN ALLOCATION

A. PRIOR LOSS OF COOLANT ACCIDENT (LOCA) MODEL ASSESSMENTS

10CFR 50.46 report dated July 22, 2008 (See Note 19)	ΔPCT = 0°F
10CFR 50.46 report dated July 20, 2009 (See Note 20)	ΔPCT = 0°F
10CFR 50.46 report dated July 20, 2010 (See Note 21)	ΔPCT = 0°F
10CFR 50.46 report dated July 18, 2011 (See Note 22)	ΔPCT = 0°F

NET PCT **PCT = 987°F**

B. CURRENT LOCA MODEL ASSESSMENTS

Radiation Heat Transfer Logic (See Note 23)	ΔPCT = 0°F
Maximum Fuel Rod Time Step Logic (See Note 24)	ΔPCT = 0°F
General Code Maintenance (NOTRUMP) (See Note 25)	ΔPCT = 0°F
Total PCT change from current assessments	Σ ΔPCT = 0°F
Cumulative PCT change from current assessments	Σ |ΔPCT| = 0°F

NET PCT **PCT = 987°F**

Attachment 1
LR-N12-0220
Peak Cladding Temperature Rack-Up Sheets

PLANT NAME: Salem Unit 2
ECCS EVALUATION MODEL: Large Break Loss of Coolant Accident (LBLOCA)
REPORT REVISION DATE: 6/19/12
CURRENT OPERATING CYCLE: 19

ANALYSIS OF RECORD (AOR)

Evaluation Model: BASH
Calculation: Westinghouse 93-PSE-G-0080, September 1993
Fuel: RFA 17 x 17
Limiting Fuel Type: RFA 17x17
Heat Flux Hot Channel Factor (F_Q) = 2.4
Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$) = 1.65
Steam Generator Tube Plugging = 25%
Limiting Break Size: Cd = 0.4
Break Location: Cold Leg
Limiting Single Failure: loss of one train ECCS flow
Reference Peak Cladding Temperature (PCT) PCT = 1978°F

MARGIN ALLOCATION

A. PRIOR LOSS OF COOLANT ACCIDENT (LOCA) MODEL ASSESSMENTS

10 CFR 50.46 report dated January 18, 1995 (See Note 4)	ΔPCT = +36°F
10 CFR 50.46 report dated December 7, 1995 (See Note 5)	ΔPCT = 0°F
10 CFR 50.46 report dated August 2, 1996 (See Note 6)	ΔPCT = 0°F
10 CFR 50.46 report dated July 11, 1997 (See Note 7)	ΔPCT = +15°F
10 CFR 50.46 report dated June 10, 1998 (See Note 8)	ΔPCT = 0°F
10 CFR 50.46 report dated April 27, 1999 (See Note 9)	ΔPCT = +24°F
10 CFR 50.46 report dated October 18, 1999 (See Note 10)	ΔPCT = -12°F
10 CFR 50.46 report dated September 21, 2000 (See Note 11)	ΔPCT = +9°F
10 CFR 50.46 report dated August 27, 2001 (See Note 12)	ΔPCT = +6°F
10 CFR 50.46 report dated August 27, 2002 (See Note 13)	ΔPCT = +20°F
10 CFR 50.46 report dated August 08, 2003 (See Note 14)	ΔPCT = +7°F
10 CFR 50.46 report dated July 29, 2004 (See Note 15)	ΔPCT = -45°F
10 CFR 50.46 report dated July 28, 2005 (See Note 16)	ΔPCT = 0°F
10 CFR 50.46 report dated July 28, 2006 (See Note 17)	ΔPCT = 0°F
10 CFR 50.46 report dated July 28, 2007 (See Note 18)	ΔPCT = +4°F
10CFR 50.46 report dated July 22, 2008 (See Note 19)	ΔPCT = -41°F
10CFR 50.46 report dated July 20, 2009 (See Note 20)	ΔPCT = 0°F
10CFR 50.46 report dated July 20, 2010 (See Note 21)	ΔPCT = 0°F
10CFR 50.46 report dated July 18, 2011 (See Note 22)	ΔPCT = 0°F

NET PCT **PCT = 2001°F**

B. CURRENT LOCA MODEL ASSESSMENTS

General Code Maintenance (BASH) (See Note 25)	$\Delta PCT = 0°F$
Total PCT change from current assessments	$\Sigma\ \Delta PCT = 0°F$
Cumulative PCT change from current assessments	$\Sigma\ \lvert\Delta PCT\rvert = 0°F$

NET PCT **PCT = 2001°F**

Attachment 2
LR-N12-0220
Assessment Notes

SALEM UNITS 1 AND 2

Docket Nos. 50-272 and 50-311

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

Report of the Emergency Core Cooling System Evaluation Model Changes and Errors Assessments

Attachment 2
LR-N12-0220
Assessment Notes

1. Prior Loss-of-Coolant Accident (LOCA) Model Assessment

The 10 CFR 50.46 report dated October 29, 1993, implemented the current Analysis of Record for the SBLOCA evaluation model (PCT = 1580°F), in support of the Fuel Upgrade / Margin Recovery Program. However, three PCT assessments were also included, resulting in a PCT benefit of -13°F. The first assessment entailed a +150°F penalty that resulted from explicitly modeling safety injection into the broken loop in the NOTRUMP model. The second assessment entailed a -150°F benefit that resulted from the implementation of an improved condensation model. The third assessment entailed a -13°F benefit that resulted from the correction of drift flux flow regime errors.

2. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated July 27, 1994, reported an assessment to the SBLOCA model, which resulted in a -16°F PCT benefit. This PCT benefit was a result of corrections made to the reactor vessel and steam generator geometric and mass calculations in the VESCAL subroutine in the LUCIFER code.

3. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated December 8, 1994, reported evaluations for the SBLOCA model due to three errors, for a penalty of +109°F. The first assessment entailed a +85°F PCT penalty that was a result of correcting nodalization and overall fluid conservation errors in the SBLOCTA code and implementing a revised transient fuel rod internal pressure model. The second assessment entailed a -6°F PCT benefit that was a result of error corrections made to the boiling heat transfer regime correlations in NOTRUMP. The third assessment entailed a +30°F PCT penalty as a result of errors affecting the steam line isolation logic in the SBLOCA evaluation model.

4. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated January 18, 1995, reported no changes in the SBLOCA model, which caused the PCT to remain unchanged. The current Analysis of Record for the LBLOCA evaluation model (PCT = 1978°F) was implemented in support of the Fuel Upgrade / Margin Recovery Program. However, three PCT assessments were also included, resulting in a PCT penalty of +36°F. The first assessment entailed a +94°F PCT penalty that resulted from the absence of Intermediate Flow Mixers (IFMs) in the core. The second assessment was a PCT benefit of -52°F that resulted from four changes to the LOCBART code; including modifications made to convert the LOCBART code from a Cray to a Unix platform, corrections made to the rod heat-up code, the addition of a new model used to determine zircaloy cladding burst behavior above 1742°F, and the implementation of a revised burst strain limit model for the rod heat-up codes. The third assessment entailed a PCT benefit of -6°F that resulted from corrections made to the LUCIFER code.

5. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated December 7, 1995, reported no changes in the SBLOCA and LBLOCA models for both Salem Units 1 and 2, which caused the PCTs to remain unchanged.

6. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated August 2, 1996, reported no changes in the LBLOCA model, which caused the PCT to remain unchanged. The SBLOCA model was assessed an -8°F PCT benefit as a result of three assessments. The first assessment was a +20°F PCT penalty due to an error in the specific enthalpy equation in NOTRUMP. The second assessment was a +10°F PCT penalty due to an error in the Fuel Rod Initialization algorithm of the SBLOCTA code, as well as several changes in the fuel rod creep and strain model. The third assessment was a -38°F PCT benefit as a result of an error in the relative loop seal elevation of the crossover leg.

7. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated July 11, 1997, reported no changes in the SBLOCA model, which caused the PCT to remain unchanged. The LBLOCA model was assessed a +15°F PCT penalty as a result of translating the fluid conditions used for subchannel analysis of the fuel rods from one computer code (SATAN) to another computer code (LOCTA).

8. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated June 10, 1998, reported no changes in the SBLOCA and LBLOCA models for both Salem Units 1 and 2, which caused the PCTs to remain unchanged.

9. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated April 27, 1999, reported no changes in the Salem Unit 1 SBLOCA and LBLOCA models, which caused the PCTs to remain unchanged. However, unit- and cycle-specific PCT assessments were applied to Salem Unit 2. For the Salem Unit 2 SBLOCA evaluation model, a generic PCT penalty of +10°F was assessed due to the impact of fully enriched annular pellets. For the Salem Unit 2 LBLOCA evaluation model, a partial re-analysis was performed that incorporated the effects of Intermediate Flow Mixers (IFMs), features of the Robust Fuel Assembly (RFA), and other model updates. The cumulative impact of these PCT changes resulted in an increase in the Salem Unit 2 LBLOCA PCT of +24°F.

10. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated October 18, 1999, reported evaluations for the SBLOCA and LBLOCA models for both Salem Units due to three errors. The first error resulted from the use of incorrect geometric data related to the accumulator lines and the pressurizer surge line. The second error was discovered in the length-averaging logic for heat transfer coefficient calculations in the LOCBART code. The third error was found in the Baker-Just metal-water reaction calculation in the LOCBART code. These errors were assessed together on a plant-specific basis and resulted in a -12°F PCT benefit for LBLOCA and no change (0°F) in the PCT for SBLOCA for both Salem Units. Thus, the Salem Unit 2 SBLOCA PCT remained unchanged, while the Salem Unit 2 LBLOCA PCT decreased by -12°F. In addition to the assessment above, further unit- and cycle-specific PCT assessments were applied to Salem Unit 1. For the Salem Unit 1 SBLOCA evaluation model, a generic PCT penalty of +10°F was assessed due to the impact of fully enriched annular pellets. For the Salem Unit 1 LBLOCA evaluation

model, a partial re-analysis was performed that incorporated the effects of the Robust Fuel Assembly (RFA) features, Intermediate Flow Mixers (IFMs), and other model updates. In addition, a generic transition core PCT penalty was assessed to account for the effects of mixed fuel types (RFA and V5H) in the core. The cumulative impact of all of these PCT changes resulted in an increase in the Salem Unit 1 LBLOCA PCT of +12°F.

11. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated September 21, 2000, reported evaluations for SBLOCA model changes, which resulted in a +27°F PCT increase. This increase consisted of a +14°F PCT assessment due to an error in the feedwater line volume calculation and a +13°F PCT assessment due to the discovery of several closely related errors dealing with mixture level tracking and region depletion errors in NOTRUMP. The LBLOCA model was assessed a +9°F PCT penalty as a result of an error in the LOCBART vapor film flow regime heat transfer correlation.

12. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated August 27, 2001, reported no changes in the SBLOCA model, which caused the PCT to remain unchanged. The LBLOCA model was assessed a +6°F PCT penalty as a result of using non-conservative cladding surface emissivity values in LOCBART.

13. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated August 27, 2002, reported no changes in the SBLOCA model, which caused the PCT to remain unchanged. The LBLOCA model was assessed a +20°F PCT penalty as a result of using a non-conservative assumption for accumulator water temperature.

14. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated August 8, 2003, reported no changes in the SBLOCA model, which caused the PCT to remain unchanged. A partial re-analysis was performed for the LBLOCA transient using the latest BASH-EM code version that incorporated the "LOCBART transient extension method," that ensured adequate termination of the fuel rod cladding temperature and oxidation transients predicted by LOCBART. This partial re-analysis allowed several prior PCT "generic evaluation" assessments (Accumulator Line / Pressurizer Surge Line Data Error, LOCBART Spacer Grid Single Phase Heat Transfer Error, LOCBART Zirc-Water Oxidation Error, LOCBART Vapor Film Flow Regime Heat Transfer Error, LOCBART Cladding Emissivity Error, Changes due to RFA Fuel Features, and Non-Conservative Accumulator Water Temperature Evaluation) to be replaced with a plant-specific analytical estimation. In addition, a +15°F PCT penalty was assessed to the LBLOCA model that resulted from corrections to the LOCBART ZIRLO Cladding Specific Heat Model. As a result of this penalty and the partial re-analysis, the LBLOCA PCT increased by +7°F.

15. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated July 29, 2004, reported a +40°F increase in the PCT of the SBLOCA evaluation model as a result of inconsistency corrections made to the NOTRUMP Bubble Rise and Drift Flux models and burst and blockage and time in life. The Salem Unit 1 LBLOCA model was assessed a

+5°F PCT penalty as a result of the correction of discrepancies in the LOCBART Fluid Property Logic. The Salem Unit 2 LBLOCA model was also assessed this +5°F penalty, in addition to the removal of a +50°F Transition Core Penalty that resulted from operating with a mixed core of V5H and RFA fuel types, for a decrease in the PCT of -45°F.

16. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated July 28, 2005, reported a 0°F increase in the PCT of the SBLOCA evaluation model due to the SBLOCA model assessment. The model assessment for SBLOCA was performed for reactor coolant pump reference conditions and general code maintenance (NOTRUMP). The report also reported a 0°F increase in the PCT of the LBLOCA evaluation model due to the LBLOCA model assessment. The model assessment for LBLOCA was performed for reactor coolant pump reference conditions, LOCBART fluid property logic, steam generator inlet/outlet plenum flow areas, initial containment relative humidity assumption and general code maintenance (BASH).

17. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated July 28, 2006, reported a 0°F increase in the PCT of the SBLOCA evaluation model due to the SBLOCA model assessment. The model assessment for SBLOCA included replacing previously transmitted pressurizer fluid volumes with nominal cold values, correcting for an error in the lower guide tube assembly weight, corrected modeling of the spilling flows in the RWST draindown calculation and general code maintenance (NOTRUMP). The report also reported a 0°F increase in the PCT of the LBLOCA evaluation model due to the LBLOCA model assessment. The model assessment for LBLOCA included replacing previously transmitted pressurizer fluid volumes with nominal cold values, correcting for an error in the lower guide tube assembly weight, and general code maintenance (BASH). Additionally, the 50°F transition core PCT penalty applied to Salem Unit 1 LBLOCA was removed.

18. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated July 25, 2007, reported a 0°F increase in the PCT of the SBLOCA evaluation model due to the SBLOCA model assessment. The model assessment for SBLOCA included the impact of the SBLOCA break size spectrum, errors in the IMP code vessel nozzle collections, and general code maintenance (NOTRUMP). The report also reported a +4°F increase in the PCT of the LBLOCA evaluation model due to the LBLOCA model assessment. The model assessment for LBLOCA included BASH minimum and maximum time step sizes (0°F), a rebaseline calculation to determine the limiting LOCBART calculated PCT (-8°F), LOCBART code correction for pellet volumetric heat generation rate (+12°F), LOCBART code option to convert user-specified zirconium-oxide thickness to equivalent cladding reacted (0°F), errors in the IMP code vessel nozzle collections (0°F), and general code maintenance (BASH).

19. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated July 22, 2008, reported a 0°F increase in the PCT of the SBLOCA evaluation model due to the SBLOCA model assessment. The model assessment for SBLOCA included the impact of errors in the reactor vessel lower plenum surface area calculation and general code

maintenance (NOTRUMP). A new Small Break LOCA Analysis of Record was implemented for Salem Unit 2 with implementation of the replacement steam generators in Salem 2 Cycle 17. The report also provided a 0°F increase in PCT of the LBLOCA evaluation model for Salem Unit 1 due to the LBLOCA model assessment. The Salem Unit 1 model assessment for LBLOCA included BASH pellet volumetric heat generation rate, error in reactor vessel lower plenum surface area calculations, and general code maintenance (BASH). The Salem Unit 2 model assessment for Large Break LOCA included a net -41°F benefit due to implementation of the replacement steam generators and change in steam generator tube plugging limits from 25% to 10% (-47°F), removal of a rebaseline calculation not applicable to Salem Unit 2 with the new steam generators (+8°F); BASH pellet volumetric heat generation rate correction (0°F); LOCBART pellet volumetric heat generation rate correction (-2°F), and errors in the reactor vessel lower plenum surface area calculation (0°F), and general code (BASH) maintenance (0°F).

20. Prior LOCA Model Assessment

The 10CFR50.46 report dated July 20, 2009, reported a 0°F increase in the PCT for the Salem Unit 1 and Salem Unit 2 small and large break LOCA model assessments. Discrepancies were discovered in the use of metal masses from drawings. The updated reactor vessel metal masses and fluid volumes have been evaluated for impact on current licensing basis analysis results and will be incorporated on a forward-fit basis. These changes represent a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451. The differences in the reactor vessel metal mass and fluid volume are relatively minor and produce a negligible effect on large and small break LOCA analysis results, leading to a PCT impact of 0°F for 10 CFR 50.46 reporting purposes. General code maintenance (NOTRUMP for SBLOCA and BASH for LBLOCA) resulted in a 0°F PCT increase for Salem Unit 1 and Salem Unit 2.

21. Prior LOCA Model Assessment

The 10CFR50.46 report dated July 20, 2010, reported a 0°F increase in the PCT for the Salem Unit 1 and Salem Unit 2 small and large break LOCA model assessments. No discrepancies were identified in the 10CFR50.46 LOCA models or methods for this reporting period for Salem Unit 1 and Salem Unit 2.

22. Prior LOCA Model Assessment

The 10CFR50.46 report dated July 18, 2011, reported a 0°F increase in the PCT for the Salem Unit 1 and Salem Unit 2 small and large break LOCA model assessments. Discrepancies were discovered and are summarized. Historically, the overall vessel average temperature uncertainty calculated by Westinghouse considered only "-" instrument uncertainties, corresponding to the indicated temperature being lower than the actual temperature. The uncertainty was then applied as a "+/-" uncertainty in some LOCA analyses, rather than using specific "+" and "-" uncertainties. This discrepancy has been evaluated for impact on existing Large and Small Break LOCA analysis results, and its resolution represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451. The issue was judged to have a negligible impact on existing Large and Small Break LOCA analysis results, leading to an estimated PCT impact of 0°F. Two issues were identified related to the normalized pellet crack and dish volumes utilized in the LOCA peak clad temperature (PCT) analyses. These issues were: 1) the incorrect tables of normalized volume versus linear heat generation rate were being used (the table for clad outer diameters of <0.4 inches were using tables for clad outer diameters >0.4 inches and vice versa), and 2) the normalized volume at 18 kw/ft was incorrectly programmed in one of the tables as

1.58 instead of 1.59. This discrepancy has been evaluated for impact on existing Large and Small Break LOCA analysis results, and its resolution represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451. These issues were judged to have a negligible impact on existing Large and Small Break LOCA analysis results, leading to an estimated PCT impact of 0°F.

23. Radiation Heat Transfer Logic

Two errors were discovered in the calculation of the radiation heat transfer coefficient in the SBLOCTA computer code. First, existing diagnostics did not preclude non-physical negative or large (negative or positive) radiation heat transfer coefficients from being calculated. These calculations occurred when the vapor temperature exceeded the cladding surface temperature or when the predicted temperature difference was less than 1 degree. Second, a temperature term incorrectly used degrees Fahrenheit instead of Rankine. These errors have been corrected in the SBLOCTA code and represent a closely related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451. A combination of SBLOCTA sensitivity calculations and engineering judgment led to an estimated PCT effect of 0°F for existing Small Break LOCA analysis results.

24. Maximum Fuel Rod Time Step Logic

An error was discovered in the SBLOCTA code that allowed the fuel rod time step to exceed the specified maximum allowable time step. The time step logic has been corrected in the SBLOCTA code. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451. A combination of SBLOCTA sensitivity calculations and engineering judgment led to an estimated PCT effect of 0°F for existing Small Break LOCA analysis results.

25. General Code Maintenance (BASH/NOTRUMP)

Various changes have been made to enhance usability and help preclude errors in analyses. This includes items such as modifying input and variable definitions, units, and defaults; improving the input diagnostic checks; enhancing the code output; optimizing active coding; and eliminating inactive coding. These changes represent Discretionary Changes that will be implemented on a forward fit basis in accordance with Section 4.1.1 of WCAP-13451. The nature of these changes leads to an estimated PCT impact of 0°F.

RISK MANAGEMENT

1. PURPOSE

1.1. This procedure specifies the requirements and responsibilities of the Risk Management (RM) Program at PSEG Nuclear facilities.

1.2. This procedure defines technical activities necessary to comply with governing regulatory requirements as they apply to the PSEG RM Program.

1.3. This procedure identifies interfaces between the RM Program and other PSEG programs and functions.

2. TERMS AND DEFINITIONS

2.1. **External Event** – An event originating external to a nuclear power plant that, in combination with safety system failures, operator errors, or both, can affect the operability of plant systems and may lead to core damage or large early radioactivity release. By convention, loss of offsite power is considered to be an internal event, and internal fire is considered to be an external event.

2.2. **Internal Event** – An event originating within a nuclear power plant that, in combination with safety system failures, operator errors, or both, can affect the operability of plant systems and may lead to core damage or large early radioactivity release. By convention, loss of offsite power is considered to be an internal event, and internal fire is considered to be an external event.

2.3. **Probabilistic Risk Analysis** – Probabilistic risk analysis (PRA) is either the method (verb) which provides risk insights or the result (noun) of applying that method, also referred to as probabilistic risk assessment, probabilistic safety analysis, or probabilistic safety assessment. These terms are all equivalent.

2.4. **PRA Model** – The PRA model is the mathematical logic structure and data which are used as a quantitative tool to conduct probabilistic risk analysis.

2.5. **Risk** – Risk is the product of the likelihood of an event during a defined period of time (usually one reactor-year) times the consequence of that event. In PRA risk is typically calculated as the frequency (conditional probability) of a core damage, radioactive release, or public health impact.

2.6. **Risk Management** – Risk management is the development, maintenance, and application of methods that provide risk insights to be used in the design, maintenance, and operation of PSEG nuclear power facilities. These methods may be either qualitative or quantitative.

2.7. **Risk Management Application** –The use of PRA models and risk management methods to formulate risk-informed input to decision processes.

3. RESPONSIBILITIES

3.1. Risk Management consists of an integrated team where individuals assigned specific roles and responsibilities delineated below will perform the actions of their position regardless of their employer and any work product will be considered a PSEG product.

3.2. **Director Engineering Services** – Provides overall governance of Risk Management as an engineering program including:

-Functional Area Manager of Risk Management program

-Approval of procedures and T&RMs as required.

3.3. **Corporate Programs Engineering Manager** – Provides management oversight of RM Program as an engineering program including:

- reviews Training and Reference Materials (T&RMs) and procedures which provide RM Program guidance
- approves RM documentation as required
- Monitors consistency of Program procedures

3.4. **Corporate Programs Engineer** – Provides technical governance, oversight and direction regarding conduct of risk management program

- Coordinates preparation and revision of Risk Management procedures and T&RMs
- Coordinates participation at industry Risk Management meetings (Owners group, EPRI etc.)
- Ensures implementation of applicable industry Risk management standards
- Approves Risk Management documentation as required
- Coordinates overall PSEG risk management resources with the Site Engineering Programs Managers

3.5. **Site Engineering Programs Manager** – Provides implementation direction for conduct of RM Program including

- assigns site risk management resources
- ensures Risk Management Engineers are properly trained and qualified as required by the PSEG training and qualification program
- provides oversight of Site Risk Management Engineer activities
- ensures compliance with this procedure
- approves RM documentation as required

- Obtains and ensures resources for site risk management activities
- Ensures timely resolution of technical issues

3.6. **Model Owner –** Conducts RM Program activities as assigned including:

- Responsibility for control and update of assigned models Maintaining content and documentation of assigned PRA models
- Completes appropriate RM training and qualification
- Ensures implementation of and compliance with applicable industry Risk management standards

3.7. **(Site) Risk Management Engineer ((S)RME)** – Conducts RM Program activities as assigned including

- completes appropriate RM training and qualification
- develops, maintains, and applies RM methods and tools
- selects appropriate RM method for any assigned application
- prepares and/or reviews RM documentation as required
- Reviews plant changes such as modifications, procedure revision and calculation revisions for impact on plant PRA models
- Facilitates training and site awareness of PRA models and Risk Management program
- Ensures compliance with Risk Management procedures and T&RMs.

4. MAIN BODY

4.1 RM Personnel Training and Qualifications

4.1.1. Training and qualification requirements shall be specified for RM Engineers.

4.1.2. RM Engineers shall comply with PSEG specified training and qualification requirements.

4.1.3. Records of RM Engineers training and qualification shall be maintained according to specified procedures or guidelines for such records.

4.2. PRA Models

4.2.1. A plant-specific, full-power, internal events PRA model (or models) shall be developed for each PSEG nuclear power unit.

4.2.2. A plant-specific, full-power, fire PRA model (or models) shall be developed for each PSEG nuclear power unit.

4.2.3. Each of the above PRA models shall be maintained reasonably representative of the as-built, as operated nuclear power unit which it represents.

4.2.4. Other PRA models representing low power or shutdown conditions or other risk sources, such as from external events, which are either plant-specific or generic may be developed.

4.2.5. The scope, level of detail, and capability of each PRA model shall be commensurate with its intended use.

4.2.6. The methodology used in each PRA model shall conform to accepted industry practices such as described in industry standards, regulatory guidelines, or other accepted guidance.

4.2.7. Each base PRA model should be evaluated at least once by available external peer review and certification processes. If a peer review and certification process is not available for a particular PRA model, that PRA model should be evaluated by best engineering practices. Periodic PRA model updates should be evaluated as required by industry standards where available.

4.3. <u>RM Applications</u>

4.3.1. Risk insights from the application of RM methods, particularly application of a PRA model, shall be used in an integrated decision making process which involves considerations of defense in depth, preservation of safety margins, and regulatory requirements.

4.3.2. If the results from application of a PRA model are used to support a docketed submittal to the NRC, the results shall comply with applicable regulatory acceptance guidelines or criteria.

4.3.3. If a PRA model is used to support any process governed by regulation, such as the Maintenance Rule (Ref. 6.3), the tools or results shall be reviewed in a specified manner by a qualified RM Engineer.

4.3.4. If a PRA model is used to support any process which impacts the risk characteristics of the plant, such as design modifications, temporary modifications, equipment safety classification, testing, inspections, and procedure changes, the results shall be reviewed in a specified manner by a qualified RM Engineer.

4.4. RM Program Interfaces

4.4.1. The RM Program supports other PSEG programs by providing tools to be used for risk assessments, criteria to be used for characterizing risk levels, expert consultation relative to interpreting risk assessment results, and interpretations of risk-informed regulations, standards, and guidelines.

4.4.2. The RM Program conducts risk management activities which are specified in other PSEG procedures (see References 6.4 through 6.8 as examples) with which the RM Program interfaces.

4.4.3. The RM Program advises the owners of interfacing PSEG programs on processes, procedures, and other interfacing program documents relative to RM topics or activities specified in those documents.

4.4.4. The RM Program shall conduct self assessments on a specified frequency and relative to specified acceptance criteria.

5. DOCUMENTATION

5.1. None

6. REFERENCES

6.1. Not used

6.2. Not used

6.3. 10CFR50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants"

6.4. CC-AA-102, "Design Input and Configuration Change Impact Screening"

6.5. CC-AA-103, "Configuration Change Control For Permanent Physical Plant Changes"

6.6. CC-AA-112, "Temporary Configuration Changes"

6.7. ER-AA-310, "Implementation of the Maintenance Rule"

6.8. WC-AA-101, "On-line Work Management Process"

7. ATTACHMENTS

7.1. None