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

PSEG
Nuclear L.L.C.

10 CFR 50.46

JUL 2 2 2008
LR-N08-0147

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 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**

REFERENCES: (1) PSEG Letter LR-N07-0181, "Salem Nuclear Generating Station Unit Nos. 1 and 2 Facility operating License DPR-70 and DPR-75 Docket Nos. 50-272 and 50-311, Annual Report for the Emergency Core Cooling System Evaluation Model Changes and Errors Required by 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," dated July 25, 2007

In accordance with 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," paragraph (2)(3)(iii), PSEG Nuclear LLC (PSEG) submits its annual report of the Emergency Core Cooling System (ECCS) Evaluation Model changes and errors for Salem Units 1 and 2. The last Peak Cladding Temperature (PCT) report PSEG Nuclear filed with the NRC for Salem was dated July 25, 2007 (Reference 1).

This letter contains two (2) enclosures. Enclosure 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. Enclosure 2, "Assessment Notes," contains a detailed description for each change or error reported.

A002
NRR

If you have any questions regarding this letter, please contact Glenn Schwartz at 856-339-1857.

Sincerely,

Robert Braun
Site Vice President Salem Station

Enclosures (2)

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

U. S. Nuclear Regulatory Commission
Attn: Mr. R. Ennis, Licensing Project Manager - Salem
Mail Stop 08B1
Washington, DC 20555-0001

USNRC Senior Resident Inspector - Salem (X24)

Mr. P. Mulligan, Manager IV
Bureau of Nuclear Engineering
P.O. Box 415
Trenton, NJ 08625

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

Peak Cladding Temperature Rack-Up Sheets

Peak Cladding Temperature Rack-Up Sheets

PLANT NAME: Salem Unit 1
ECCS EVALUATION MODEL: Small Break Loss of Coolant Accident (SBLOCA)
REPORT REVISION DATE: 6/23/2008
CURRENT OPERATING CYCLE: 19

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

NET PCT **PCT = 1729°F**

B. CURRENT LOCA MODEL ASSESSMENTS

Error in Reactor Vessel Lower Plenum Surface Area Calculation (Note 23)	ΔPCT = 0°F
General Code Maintenance (NOTRUMP) (Note 24)	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**

Peak Cladding Temperature Rack-Up Sheet

PLANT NAME: Salem Unit 1
ECCS EVALUATION MODEL: Large Break Loss of Coolant Accident (LBLOCA)
REPORT REVISION DATE: 6/23/2008
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 = 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 (Note 4)	ΔPCT = +36°F
10 CFR 50.46 report dated December 7, 1995 (Note 5)	ΔPCT = 0°F
10 CFR 50.46 report dated August 2, 1996 (Note 6)	ΔPCT = 0°F
10 CFR 50.46 report dated July 11, 1997 (Note 7)	ΔPCT = +15°F
10 CFR 50.46 report dated June 10, 1998 (Note 8)	ΔPCT = 0°F
10 CFR 50.46 report dated April 27, 1999 (Note 9)	ΔPCT = 0°F
10 CFR 50.46 report dated October 18, 1999 (Note 10)	ΔPCT = +12°F
10 CFR 50.46 report dated September 21, 2000 (Note 11)	ΔPCT = +9°F
10 CFR 50.46 report dated August 27, 2001 (Note 12)	ΔPCT = +6°F
10 CFR 50.46 report dated August 27, 2002 (Note 13)	ΔPCT = +20°F
10 CFR 50.46 report dated August 08, 2003 (Note 14)	ΔPCT = +7°F
10 CFR 50.46 report dated July 29, 2004 (Note 15)	ΔPCT = +5°F
10 CFR 50.46 report dated July 28, 2005 (Note 16)	ΔPCT = 0 °F
10 CFR 50.46 report dated July 28, 2006 (Note 17)	ΔPCT = -50 °F
10 CFR50.46 report dated July 25, 2007 (Note 18)	ΔPCT = +4°F

NET PCT **PCT = 2042°F**

B. CURRENT LOCA MODEL ASSESSMENTS

BASH Pellet Volumetric Heat Generation Rate (Note 19)	$\Delta PCT = 0°F$
Error in Reactor Vessel Lower Plenum Surface Area Calculation (Note 23)	$\Delta PCT = 0°F$
General Code Maintenance (BASH Code) (Note 24)	$\Delta PCT = 0°F$
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 = 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: 6/23/2008
CURRENT OPERATING CYCLE: 17*

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 = 25%
Limiting Break Size: 2 inches
Break Location: Cold Leg
Single Failure: loss of one train 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 (Note 1)	ΔPCT = -13°F
10 CFR 50.46 report dated July 27, 1994 (Note 2)	ΔPCT = -16°F
10 CFR 50.46 report dated December 8, 1994 (Note 3)	ΔPCT = +109°F
10 CFR 50.46 report dated January 18, 1995 (Note 4)	ΔPCT = 0°F
10 CFR 50.46 report dated December 7, 1995 (Note 5)	ΔPCT = 0°F
10 CFR 50.46 report dated August 2, 1996 (Note 6)	ΔPCT = -8°F
10 CFR 50.46 report dated July 11, 1997 (Note 7)	ΔPCT = 0°F
10 CFR 50.46 report dated June 10, 1998 (Note 8)	ΔPCT = 0°F
10 CFR 50.46 report dated April 27, 1999 (Note 9)	ΔPCT = +10°F
10 CFR 50.46 report dated October 18, 1999 (Note 10)	ΔPCT = 0°F
10 CFR 50.46 report dated September 21, 2000 (Note 11)	ΔPCT = +27°F
10 CFR 50.46 report dated August 27, 2001 (Note 12)	ΔPCT = 0°F
10 CFR 50.46 report dated August 27, 2002 (Note 13)	ΔPCT = 0°F
10 CFR 50.46 report dated August 08, 2003 (Note 14)	ΔPCT = 0°F
10 CFR 50.46 report dated July 29, 2004 (Note 15)	ΔPCT = +40°F
10 CFR 50.46 report dated July 28, 2005 (Note 16)	ΔPCT = 0°F
10 CFR 50.46 report dated July 28, 2006 (Note 17)	ΔPCT = 0°F
10 CFR 50.46 report dated July 28, 2007 (Note 18)	ΔPCT = 0°F

NET PCT **PCT = 1729°F**

B. CURRENT LOCA MODEL ASSESSMENTS

Error in Reactor Vessel Lower Plenum Surface Area Calculation (Note 23)	ΔPCT = 0°F
General Code Maintenance (NOTRUMP) (Note 24)	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**

* Note: Since the last annual report, Salem Unit 2 completed operating cycle 16 on March 11, 2008. During the refueling outage, the steam generators were replaced. This PCT rack-up reflects operating with the old steam generators. SBLOCA impacts associated with the replacement steam generators are provided in a separate PCT rack-up sheet.

Peak Cladding Temperature Rack-Up Sheets

PLANT NAME: Salem Unit 2 (with replacement steam generators)
ECCS EVALUATION MODEL: Small Break Loss of Coolant Accident (SBLOCA)
REPORT REVISION DATE: 6/23/2008
CURRENT OPERATING CYCLE: 17

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

None (Note 20)	ΔPCT = 0°F

NET PCT **PCT = 987°F**

B. CURRENT LOCA MODEL ASSESSMENTS

Error in Reactor Vessel Lower Plenum Surface Area Calculation (Note 23)	ΔPCT = 0°F
General Code Maintenance (NOTRUMP) (Note 24)	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**

Peak Cladding Temperature Rack-Up Sheets

PLANT NAME: Salem Unit 2
ECCS EVALUATION MODEL: Large Break Loss of Coolant Accident (LBLOCA)
REPORT REVISION DATE: 6/23/2008
CURRENT OPERATING CYCLE: 17*

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

NET PCT **PCT = 2042°F**

B. CURRENT LOCA MODEL ASSESSMENTS

BASH Pellet Volumetric Heat Generation Rate (Note 19)	ΔPCT = 0°F
Error in Reactor Vessel Lower Plenum Surface Area Calculation (Note 23)	ΔPCT = 0°F
General Code Maintenance (BASH Code) (Note 24)	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 = 2042°F**

* Note: Since the last annual report, Salem Unit 2 completed operating cycle 16 on March 11, 2008. During the refueling outage, the steam generators were replaced. This PCT rack-up reflects operating with the old steam generators. LBLOCA impacts associated with the replacement steam generators are provided in a separate PCT rack-up sheet.

Peak Cladding Temperature Rack-Up Sheets

PLANT NAME: Salem Unit 2 (with replacement steam generators)
ECCS EVALUATION MODEL: Large Break Loss of Coolant Accident (LBLOCA)
REPORT REVISION DATE: 6/23/2008
CURRENT OPERATING CYCLE: 17

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

NET PCT **PCT = 2042°F**

B. CURRENT LOCA MODEL ASSESSMENTS

BASH Pellet Volumetric Heat Generation Rate (Note 19)	ΔPCT = 0°F
LOCBART Pellet Volumetric Heat Generation Rate (Note 21)	ΔPCT = -2°F
Changes Associated with Replacement Steam Generators (Note 22)	ΔPCT = +8°F
Error in Reactor Vessel Lower Plenum Surface Area Calculation (Note 23)	ΔPCT = 0°F
General Code Maintenance (BASH Code) (Note 24)	ΔPCT = 0°F
Total PCT change from current assessments	Σ ΔPCT = +6°F
Cumulative PCT change from current assessments	Σ |ΔPCT| = +10 °F

C. 10 CFR 50.59 CHANGES

Replacement Steam Generator Evaluation (Note 25)	ΔPCT = -47°F

NET PCT **PCT = 2001°F**

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

Assessment Notes

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. BASH Pellet Volumetric Heat Generation Rate

The Bash code has been modified to correct an inverted term in the calculation of the pellet volumetric heat generation rate. This change affects the steady-state and transient heat generation for the core average rod prior to bottom-of-core recovery and could result in either an increase or decrease in the cladding temperature at the beginning of reflood. The change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Sensitivity calculations using BASH and SMUUTH indicated a negligible effect on the core inlet flooding rate during reflood, leading to an estimated impact of 0°F for 10CFR50.46 reporting purposes.

20. New Salem Unit 2 SBLOCA Analysis of Record

Salem Unit 2 License Amendment 267 implemented WCAP-10054-P-A, Addendum 2, Revision 1 "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection in the Broken Loop and COSI Condensation Model". This results in a new SBLOCA analysis of record applicable only to Salem Unit 2 with the replacement steam generators (RSGs), starting with Cycle 17.

21. LOCBART Pellet Volumetric Heat Generation for Salem Unit 2 with Replacement Steam Generators

As described in the 2007 annual report, the LOCBART code was modified to correct an inverted term in the calculation of the pellet volumetric heat generation rate. This change affects the steady-state and transient heat generation that could result in either an increase or decrease in peak cladding temperature for a given calculation. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451. As reported in 2007 this assessment was estimated to be +12°F. For the Salem Unit 2 with RSGs the assessment was estimated to be +10°F. Thus the effect of this change from the prior LBLOCA rack-up is -2°F.

22. LBLOCA changes associated with Salem 2 with Replacement Steam Generators

The Salem 2 LBLOCA rack-up sheet provided in the 2007 annual report included a -8°F PCT benefit from a rebaseline of the limiting LOCBART calculation. With the RSGs this rebaseline assessment no longer applies. Therefore +8°F was added as a new assessment to cancel the effect of the prior rebaseline calculation.

23. Error in Reactor Vessel Lower Plenum Surface Area Calculations

Two errors were discovered in the calculations of reactor vessel lower plenum surface area. The corrected values were 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 vessel lower plenum surface area are relatively minor and would be expected to produce a negligible effect on large and small break LOCA analysis results, leading to an estimated PCT impact of 0°F for 10 CFR 50.46 reporting purposes. The affected Westinghouse evaluation models include 1981 BASH (Large Break LOCA) and 1985 NOTRUMP (Small Break LOCA).

24. 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.

25. Salem Unit 2 Planned Plant Modification LBLOCA Evaluation

Westinghouse evaluated the effect of the Salem Unit 2 RSG as a plant modification (not as a change to the evaluation model) per Sections 2.2 and 3.5 of WCAP-13451. The impact of the RSGs on the Salem design basis has been reviewed per 10 CFR 50.59. The evaluation has shown that no design or regulatory limit related to LBLOCA would be exceeded due to the RSG. The LBLOCA transient has minor dependency on the physical changes associated with the steam generator. However, by modeling a lower steam generator tube plugging (reduced from 25% to 10%), the RSG experiences less primary side hydraulic resistance compared to the original steam generator (OSG) resulting in a -47°F PCT benefit.