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

PSEG
Nuclear LLC

SEP 26 2006

LR-N06-0213
LCR S06-02

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATIONS TO REVISE TOPICAL REPORT REFERENCES IN TECHNICAL SPECIFICATION 6.9.1.9, CORE OPERATING LIMITS REPORT
SALEM GENERATING STATION UNITS 1 AND 2
DOCKET NOS. 50-272 AND 50-311
FACILITY OPERATING LICENSE NOS. DPR-70 AND DPR-75

References:

1) Letter from Robert C. Jones (U.S. NRC) to N. J. Liparulo (Westinghouse Electric Corporation) dated August 12, 1996, "WCAP-10054-P, Addendum 2, Revision 1, 'NOTRUMP SBLOCA Using the COSI Steam Condensation Model,' (TAC NO. M90784)."
2) Letter from Victor Nerses (U.S. NRC) to David A. Christian (Dominion Nuclear Connecticut, Inc.) dated March 9, 2004, "Millstone Power Station, Unit No. 3 – Issuance of Amendment Re: Relocation of Technical Specification Parameters to the Core Operating Limits Report (TAC NO. MB8387)."

In accordance with the provisions of 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby transmits a request for amendment of the Technical Specifications (TS) for the Salem Generating Station Units 1 and 2. Pursuant to the requirements of 10CFR50.91(b)(1), a copy of this request for amendment has been sent to the State of New Jersey.

The proposed amendments would revise Core Operating Limits Report (COLR) topical report references in TS 6.9.1.9.b for Salem Unit 1 and 2, and, for Salem Unit 2, add a reference to WCAP-10054-P-A Addendum 2, a Small Break Loss of Coolant Accident (SBLOCA) topical report that was generically approved by the NRC staff via Reference 1. The proposed changes are consistent with NRC-approved Revision 0 to Technical Specification Task Force (TSTF) Traveler 363, "Revise Topical Report References in ITS 5.6.5, COLR," and are similar to changes approved in Amendment No. 218 to the Millstone Unit 3 Operating License (Reference 2).

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PSEG has evaluated the proposed changes in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and has determined this request involves no significant hazards considerations. An evaluation of the requested changes is provided in Attachment 1. The marked up TS affected by the proposed change is provided in Attachment 2.

Approval of this change is requested by August 31, 2007, with Salem Unit 1 implementation prior to restart from its 19th refueling outage in Fall, 2008, and Salem Unit 2 implementation prior to restart from its sixteenth refueling outage in Spring, 2008.

Should you have any questions regarding this request, please contact Mr. James Mallon at 610-765-5507.

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

Sincerely,

Executed on 9/25/06

Thomas P. Joyce
Site Vice President
Salem Station Units 1 and 2

Attachments (2)

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

Mr. Stewart Bailey, Licensing Project Manager
U. S. Nuclear Regulatory Commission
Mail Stop 08B1
Washington, DC 20555-0001

Ms. Farideh Saba, Licensing Project Manager
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USNRC Senior Resident Inspector - Salem (X24)

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

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

REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATIONS TO REVISE TOPICAL REPORT REFERENCES IN TECHNICAL SPECIFICATION 6.9.1.9, CORE OPERATING LIMITS REPORT

Table of Contents

CHANGES TO TECHNICAL SPECIFICATIONS

1.0 DESCRIPTION

The purpose of this amendment request is to revise the list of NRC-approved methodologies in Technical Specification (TS) 6.9.1.9.b that are used to develop core operating limits. The proposed changes are to remove the revision number and date for the referenced topical reports based on identification of specific NRC-approved revision levels in the Core Operating Limits Report (COLR).

For Salem Unit 2, the proposed change also adds a reference to WCAP-10054-P-A, Addendum 2, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model" for Small Break Loss of Coolant Accident (SBLOCA) analyses, revision 1 of which was generically approved by NRC as documented in Reference 1.

2.0 PROPOSED CHANGE

Technical Specification (TS) 6.9.1.9.b lists applicable references for the analytical methods used to determine core operating limits identified in TS 6.9.1.9.a. The proposed change would remove the revision level and date for each of the topical reports listed in TS 6.9.1.9.b.

For Salem Unit 2, the proposed change would also add the following as reference 7 to TS 6.9.1.9.b:

"7. WCAP-10054-P-A, Addendum 2, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model."

3.0 BACKGROUND

3.1 Removal of Revision Levels and Dates from TS 6.9.1.9.b References

Technical Specification (TS) 6.9.1.9 requires that the analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC; TS 6.9.1.9.b lists the NRC-approved methods for determining core operating limits. Technical Specification Task Force (TSTF) Traveler 363, "Revise Topical Report References in ITS 5.6.5, COLR," Revision 0, endorses the removal of revision levels and dates for the NRC-approved topical reports referenced in TS 6.9.1.9.b, provided the specific revision levels are identified in the Core Operating Limits Report (COLR) submitted to NRC upon issuance for each reload cycle. This method of referencing topical reports enables licensees to use current topical reports to support limits in the COLR without having to submit an amendment request each time the topical report is revised. This eliminates the unnecessary expenditure of NRC and licensee resources associated with processing license amendments, while continuing to assure that NRC-approved methods are used to determine core operating limits.

3.2 Addition of NOTRUMP Topical Report Addendum to TS 6.9.1.9.b

Westinghouse topical report WCAP-10054-P-A, Revision 1 is referenced in TS 6.9.1.9.b.3 and describes the Small Break Loss of Coolant Accident (SBLOCA) Emergency Core Cooling System (ECCS) evaluation model used for Salem Unit 1 and 2. Addendum 2 to WCAP-10054 (Reference 1) describes the COSI steam condensation model for use in the NOTRUMP SBLOCA analyses.

In support of the Salem Unit 2 steam generator replacement planned for the Spring 2008 refueling outage, Westinghouse reanalyzed the SBLOCA, modeling the Replacement Steam Generators (RSGs) and using the COSI model consistent with Reference 1. Because the COSI model represents a change in ECCS evaluation methods, PSEG proposes to add Addendum 2 to WCAP-10054 to Salem Unit 2 TS 6.9.1.9.b.

The COSI model provides increased condensation efficiencies for Safety Injection (SI) flows to the faulted and intact Reactor Coolant System (RCS) loops, thereby resulting in lower calculated Peak Clad Temperature (PCT).

Reference 1 includes the NRC's August 12, 1996 safety evaluation that generically approves the NOTRUMP COSI model. As described below in Section 4, application of the COSI model to the Salem Generating Station complies with the conditions of NRC's generic approval.

4.0 TECHNICAL ANALYSIS

4.1 Removal of Revision Levels and Dates from TS 6.9.1.9.b References

Technical Specification Task Force (TSTF) Traveler 363 endorses the removal from TS of revision levels and dates of the NRC-approved methods for determining core operating limits. TS 6.9.1.9.b continues to require that the analytical methods used to determine core operating limits are reviewed and approved by NRC. The Core Operating Limits Report (COLR), including any mid-cycle revisions or supplements, is submitted to NRC upon issuance. The Salem COLR currently references TS 6.9.1.9.b rather than separately identifying the analytical methods used. As part of implementation of the requested amendment, the COLR format will be revised to specifically list the NRC-approved methods, including dates and revision levels. Future changes to COLR methodology revision levels will be subject to 10CFR50.59, thereby ensuring plant-specific application of the revision either provides results that are essentially the same, or is approved by NRC for the intended application (i.e., that the conditions of approval are met for the specific application). The proposed change would therefore continue to provide assurance that core operating limits are determined using NRC-approved methods.

4.2 Addition of NOTRUMP Topical Report Addendum to TS 6.9.1.9.b

The small break LOCA (SBLOCA) analysis of record for Salem Unit 1 and 2 has been completed using the 1985 Westinghouse SBLOCA Evaluation Model with NOTRUMP (NOTRUMP-EM) referenced in TS 6.9.1.9.b.3. The SBLOCA analysis that incorporated the Salem Unit 2 Replacement Steam Generator (RSG) design includes the addition of the COSI model (Reference 1), which is a change in the methodology from the previous SBLOCA analysis-of-record.

This SBLOCA reanalysis was performed to incorporate the RSG design. The small break LOCA analysis uses plant-specific parameters that are bounded by the models and correlations contained in the generic methodology. Therefore, the Salem Unit 2 analysis conforms to 10 CFR 50.46 and Section II of Appendix K.

The conclusions of the analysis are that the criteria of 10 CFR 50.46(b) are met:

1. The calculated peak cladding temperature (PCT) resulting from SBLOCA reanalysis using COSI condensation model is 987°F, well below the limit of 2200°F.
2. The maximum local oxidation is 0.01%, vs. the 10 CFR 50.46(b) limit of 17%.
3. The core-wide hydrogen generation remains well below the 10 CFR 50.46 acceptance limit of 1 percent.
4. The core geometry remains amenable to cooling.
5. After successful initial operation of the ECCS, the calculated core temperature will be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The current Large Break LOCA (LBLOCA) peak cladding temperature is 2038°F according to Reference 2. Therefore, due to the low SBLOCA peak cladding temperature and oxidation results, and the fact that the SBLOCA results are significantly non-limiting when compared with the current LBLOCA results, the standard "integer" break spectrum was used, and a refined break spectrum (i.e., break size intervals on the order of 0.25 inch) was not considered in the analysis.

Reference 1 includes the August 12, 1996 NRC safety evaluation of the COSI model that concludes that the proposed correlation for Safety Injection (SI) is a conservative representation of the condensation process during ECCS operation, and the proposed model as documented in Reference 1 was found acceptable. Specific limitations of NRC approval in the August 12, 1996 safety evaluation are

i. the range of injection jet velocities used in the experiments bracketed the corresponding rates in small break LOCAs for Westinghouse plants and the model will be used within the experimental range.

ii. Westinghouse submitted analyses demonstrating the condensation efficiency is virtually independent of Reactor Coolant System (RCS) pressure and stated the COSI model will be applied within the pressure range of 550 to 1200 psia.

Salem Unit 2 RSG SBLOCA analysis complies with the above limitations. Internal guidance used by the Westinghouse analysts provides assurance that the SBLOCA analysis satisfies all restrictions and/or requirements imposed by the Nuclear Regulatory Commission (NRC). PSEG and Westinghouse have ongoing processes to assure that the values and ranges of the SBLOCA analyses parameter inputs conservatively bound the values and ranges of the as-operated plant for those parameters. Following implementation of the proposed change, SBLOCA analyses will continue to demonstrate compliance to 10 CFR 50.46 using NRC approved methods and plant-specific input assumptions consistent with the conditions of NRC approval of the analytical methods.

Therefore, implementation of the proposed changes to add the topical report for the SBLOCA COSI condensation model in conjunction with the small break ECCS evaluation model using the NOTRUMP Code for Salem Unit 2 would not adversely affect the health and safety of the public.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

PSEG has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below.

The changes that are being evaluated are the removal of the revision levels and dates of the NRC-approved analytical methods for determining core operating limits in Salem Unit 1 and 2 Technical Specification (TS) 6.9.1.9.b; and the addition of WCAP-10054-P-A, Addendum 2, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," to Salem Unit 2 TS 6.9.1.9.b.

1. **Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed changes affect the administrative controls section of Technical Specifications (TS) that govern the analytical methods used to determine core operating limits. Removal of revision levels and dates from NRC-approved methods listed in TS is an administrative change that has no impact on the probability or consequences of an accident. TS 6.9.1.9.b will still require these methods to be reviewed and approved by NRC. The proposed change does not affect the required TS actions to be taken in the event that any core operating limits are exceeded.

   The proposed use of WCAP-10054-P-A, Addendum 2 for the Salem Unit 2 Small Break Loss of Coolant Accident (SBLOCA) analysis is consistent with the limitations and conditions of NRC approval. The parameters assumed in the analysis are within the design limits of the plant equipment. Therefore, there will be no increase in the probability of a loss of coolant accident. The consequences of a LOCA are not being increased, since it is shown that the Emergency Core Cooling System (ECCS) is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, Paragraph b. No other accident is potentially affected by this change.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No

   No new modes of plant operation are being introduced. The parameters assumed in the analysis are within the design limits of the plant equipment. TS will continue to require operation within the core operating limits determined using NRC-approved analytical methods and the proposed change does not affect any actions required in the event the core operating limits are exceeded.

   Therefore, the proposed change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   The proposed changes do not have any impact on plant equipment or safety analysis acceptance criteria. Core operating limits will continue to be determined using NRC-approved analytical methods. The ECCS acceptance criteria of 10 CFR 50.46 will continue to be met following the proposed use of WCAP-10054-P-A, Addendum 2 for the Salem Unit 2 SBLOCA analysis

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

5.2 Applicable Regulatory Requirements/Criteria

As specified in 10 CFR 50.36(c)(5), administrative controls contained in Technical Specifications "are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." The proposed changes only affect the administrative controls section of TS and are consistent with 10 CFR 50.36. The TS will continue to require the use of NRC-approved methods to define core operating limits.

10 CFR 50.46, Acceptance Criteria For Emergency Core Cooling Systems For Light-Water Nuclear Power Reactors, requires ECCS cooling performance to be calculated in accordance with an acceptable evaluation model. Section 4 of this analysis demonstrates that the proposed change is consistent with 10 CFR 50.46.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. PSEG has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22 (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," Thompson, C. M., et al., July 1997.

2. PSEG Letter to NRC LR-N06-0331, Annual Report of the Emergency Core Cooling System Evaluation Models Changes and Errors required by 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." Salem Nuclear Generating Station Unit Nos. 1 and 2, dated July 28, 2006.

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating Licenses DPR-70 and DPR-75 are affected by this change request:

Salem Unit 1

Technical Specification	**Pages**
6.9.1.9	**6-24** **6-24a**

Salem Unit 2

Technical Specification	**Pages**
6.9.1.9	**6-24** **6-24a**

6.9.1.9 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Moderator Temperature Coefficient Beginning of Life (BOL) and End of Life (EOL) limits and 300 ppm surveillance limit for Specification 3/4.1.1.4,

2. Control Bank Insertion Limits for Specification 3/4.1.3.5,

3. Axial Flux Difference Limits and target band for Specification 3/4.2.1,

4. Heat Flux Hot Channel Factor, F_Q, its variation with core height, K(z), and Power Factor Multiplier PF_{xy}, Specification 3/4.2.2, and

5. Nuclear Enthalpy Hot Channel Factor, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

6. Refueling boron concentration per Specification 3.9.1

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

Delete

1. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, July 1985 (W Proprietary), Methodology for Specifications listed in 6.9.1.9.a Approved by Safety Evaluation dated May 28, 1985. Delete

2. WCAP-8385, Power Distribution Control and Load Following Procedures - Topical Report, September 1974 (W Proprietary) Methodology for Specification 3/4.2.1 Axial Flux Difference. Approved by Safety Evaluation dated January 31, 1978.

3. WCAP-10054-P-A, Rev. 1, Westinghouse Small Break ECCS Evaluation Model Using NOTRUMP Code, August 1985 (W Proprietary), Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor. Approved for Salem by NRC letter dated August 25, 1993.

4. WCAP-10266-P-A, Rev. 2, The 1981 Version of Westinghouse Evaluation Model Using BASH Code, Rev. 2. March 1987 (W Proprietary) Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor. Approved by Safety Evaluation dated November 13, 1986.

5. WCAP-12472-P-A, BEACON - Core Monitoring and Operations Support System, Revision 0, (W Proprietary). Approved February 1994.

6. CENPD-397-P-A, Rev. 1, Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology, May 2000.

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c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

6.9.1.10 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with the Specification 6.8.4.i, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date, and

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

6.9.1.9 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Moderator Temperature Coefficient Beginning of Life (BOL) and End of Life (EOL) limits and 300 ppm surveillance limit for Specification 3/4.1.1.4,

2. Control Bank Insertion Limits for Specification 3/4.1.3.5,

3. Axial Flux Difference Limits and target band for Specification 3/4.2.1,

4. Heat Flux Hot Channel Factor, F_Q, its variation with core height, K(z), and Power Factor Multiplier PF_{xy}, Specification 3/4.2.2, and

5. Nuclear Enthalpy Hot Channel Factor, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

6. Refueling boron concentration per Specification 3.9.1

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

Delete

1. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, July 1985 (W Proprietary), Methodology for Specifications listed in 6.9.1.9.a. Approved by Safety Evaluation dated May 28, 1985.

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6. CENPD-397-P-A, Rev. 1, Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology, May 2000 Delete

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c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements shall be provided upon issuance for each reload cycle to the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Updated Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U. S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator of the Regional Office of the NRC via the Licensee Event Report System within 30 days.

6.9.4 When a report is required by ACTION 8 OR 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and the plans and schedule for restoring the instrument channels to OPERABLE status.

Insert 1 to Salem Unit 2 Technical Specification Page 6-24a

7. WCAP-10054-P-A, Addendum 2, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model."