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May 21, 2003

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

ATTENTION: Chief, Information Management Branch
Division of Program Management
Policy Development and Analysis Staff

Subject: Duke Energy Corporation
Oconee Nuclear Station - Units 1, 2, and 3
Docket Nos. 50-269, 50-270, and 50-287

Revisions to Topical Reports DPC-NE-3000 and 3005 In Support of Steam Generator Replacement
Response to NRC Staff Request for Additional Information

Reference: Duke Submittal Dated June 13, 2002

Enclosed herein, please find the Duke Energy Corporation (Duke) response to the September 24, 2002 NRC staff's request for additional information concerning topical reports DPC-NE-3000, Revision 3, "Thermal-Hydraulic Transient Analysis Methodology," and DPC-NE-3005, Revision 2, "UFSAR Chapter 15 Transient Analysis Methodology."

Please note that there is information enclosed which Duke considers proprietary. In accordance with 10CFR2.790, Duke requests that this information be withheld from public disclosure. An affidavit attesting to the proprietary nature of this information is included in this letter.

Individual questions contained in the request for additional information were discussed in a conference call with the NRC staff on October 3, 2002. Attachment 1 (proprietary version) and Attachment 2 (non-proprietary version) to this letter constitute Duke's response to those questions. Additional

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information requested by the NRC staff during the conference call has also been included.

During the completion of the analyses using the revised methodology, some additional revisions to the methodology were identified as necessary. These new revisions are detailed in Attachments 3 (proprietary version) and 4 (non-proprietary version) for topical report DPC-NE-3000, Revision 3, and in Attachments 5 (proprietary version) and 6 (non-proprietary version) for topical report DPC-NE-3005, Revision 2. Duke requests that the NRC include these revisions within the scope of review of the original June 13, 2002 submittal.

If there are any questions or if additional information is needed on this matter, please call J. A. Effinger at (704) 382-8688.

Very truly yours,

TS Canady

K. S. Canady

xc:

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## AFFIDAVIT

1) I am a Vice President of Duke Energy Corporation; and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant licensing; and I am authorized on the part of said Corporation (Duke) to apply for this withholding.

2) I am making this affidavit in conformance with the provisions of 10CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding, which accompanies this affidavit.

3) I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4) Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld:

   a) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

   b) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs relative to a method of analysis that provides a competitive advantage to Duke.

   c) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, it is to be received in confidence by the NRC.

   d) The information sought to be protected is not available in public to the best of our knowledge and belief.

K. S. Canady

(Continued)

e) The proprietary information sought to be withheld in this submittal is that which is that which is marked in Attachments 1, 3, and 5 to Duke Energy Corporation letter dated May 21, 2003; Subject: Revisions to Topical Reports DPC-NE-3000 and 3005 in Support of Steam Generator Replacement; Response to NRC Staff Request for Additional Information. This information enables Duke to:

   i) Respond to NRC requests for information regarding transient response of Babcock & Wilcox Pressurized Water Reactors.

   ii) Simulate UFSAR Chapter 15 transients and accidents for Oconee Nuclear Station.

   iii) Perform safety evaluations per 10CFR50.59.

   iv) Support Facility Operating License/Technical Specifications amendments for Oconee Nuclear Station.

f) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke. For example, it minimizes vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants. In addition, it provides increased flexibility in the implementation of changes to or evaluating conditions at Duke's nuclear plants.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

K. S. Canady

(Continued)

K. S. Canady affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

K S Canady

K. S. Canady, Vice President

Subscribed and sworn to on this 21st day of

May, 2003

Mary P. Nelms

Notary Public

My Commission Expires:

JAN 22, 2006

SEAL

bxc:

R. O. Sharpe
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ELL

## Attachment 2

## Duke Power Response To
## Request for Additional Information Regarding Topical Reports
## DPC-NE-3000 Revision 3
## "Thermal-Hydraulic Transient Analysis Methodology"
## and
## DPC-NE-3005, Revision 2
## "UFSAR Chapter 15 Transient and Accident Methodology"

The following questions relate to proposed changes to DPC-NE-3000 Revision 2, that will be incorporated to produce new DPC-NE-3000 Revision 3.

**Questions 1-8 relate to proposed changes to DPC-NE-3000-PA Revision 2, that will be incorporated to produce new DPC-NE-3000-P Revision 3.**

1. Page 1-6 states that "Post-trip decay heat is calculated with the built-in ANS Standard 1979 decay heat option." The RETRAN-3D code does not include the contribution from neutron capture within stable fission products that is part of the standard. Discuss how the multiplier to account for neutron capture within stable isotopes will be included in the RETRAN-3D input model for Oconee.

Response: The ANS-5.1-1979 standard, including the "G(t)" factor to account for the effect of neutron capture in fission products, is coded in a separate FORTRAN application. All of the Oconee-specific data required by the standard are input to this application to obtain a bounding decay heat vs. time result including 2$\sigma$ uncertainty. A multiplier vs. time is then determined and is input to RETRAN-3D via the "KMUL" variable on Card 146000. This multiplier is confirmed to produce conservatively bounding decay heat vs. time in conjunction with the RETRAN-3D 1979 decay heat model by comparing with the results from the stand-alone FORTRAN decay heat application. In this manner the effect of neutron capture in fission products is accounted for consistent with the ANS standard.

2. Page 1-7 states the pressurizer model will include the use of [ ] to determine the heat transfer and that this calculation will increase the accuracy of the model. The developers of RETRAN-3D have provided a non-equilibrium pressurizer option which could also provide additional accuracy by modeling the additional pressurizer pressure which would be obtained from the superheating of the vapor space that would occur following insurges. Discuss the basis for not using this option.

Response: Duke has used the non-equilibrium pressurizer model in RETRAN-02, and we will continue to use the non-equilibrium pressurizer model in RETRAN-3D. Refer to Section 2.2.1.4 of topical report DPC-NE-3000.

3. Page 1-8 discusses a model revision by which post-trip decay heat is calculated with the built-in 1979 ANS Standard for McGuire/Catawba transient analysis. The 1979 ANS Standard is built into RETRAN-3D. Will McGuire and Catawba be analyzed using RETRAN-3D rather than RETRAN-02? If so, please provide sections discussing RETRAN-3D modeling for McGuire/Catawba. We note that other discussions in your submittal including the evaluation of the RETRAN-3D SER conditions and limitations are only for Oconee.

Response: This change is only intended to update the decay heat modeling for McGuire/Catawba to what is currently done using RETRAN-02. Duke is not proposing to use RETRAN-3D for McGuire/Catawba with this revision.

Note: The RAI included two question #3's.

3. Figure 2.1-6 of DPC-NE-3000-PA illustrates an Oconee Once-Through Steam Generator. Will a figure be added illustrating an Oconee ROTSG? If not, why not?

Response: Refer to Figure B-1 on p. 1-17 of Attachment 1 to the June 13, 2002 submittal. This figure will be included in Appendix B when DPC-NE-3000-PA and -A, Revision 3 is published.

4. Page 1-11 of the proposed Appendix B to DPC-NE-3000-P states that one difference between the ROTSGs and the current OTSGs is that more water is contained in each generator. Please provide the water mass contained in the replacement steam generators at full power and hot standby. Discuss the physical changes in the steam generator design which cause the increase in water mass.

Response: The water mass in the OTSGs and the ROTSGs is determined by the physical dimensions, the setting of the adjustable orifice plate near the bottom of the downcomer, and how much boiling height in the tube bundle is necessary to transfer the heat load at the specified secondary pressure. Relative to the OTSGs, the downcomer is wider due to the thinner pressure vessel, there are more tubes, the Inconel-690 tubes have a lower thermal conductivity, and the pressure is higher. The adjustable orifice plate will be set to obtain a target operate range (OR) level of between 60-75% OR. The full power liquid mass for the ROTSGs at 75% OR level is approximately [ ] lbm. The water mass at hot standby (30 inches startup range) is approximately [ ] lbm.

5. Please provide any changes in steam generator level setpoints and design superheat associated with the replacement steam generators.

Response: There are no changes in the replacement steam generator level setpoints relative to the original steam generators. The design superheat is predicted to be 59°F.

6. Chapter 4.0 of DPC-NE-3000-PA describes benchmarks of the Oconee RETRAN-02 model against nine plant transients. Will similar benchmarks be performed for the RETRAN-3D model using the new steam generator slip formulations based on plant data

from startup testing following installation of the ROTSGs on Oconee Unit 1? If not, why not?

Response: Duke is not planning to conduct any transient testing following replacement steam generator installation. The replacement steam generators are very similar in design to the original steam generators, and no significant differences in response to operational transients exist. The RETRAN-3D model has been benchmarked to the BWC steam generator design code for steady-state conditions. During the first startup evolution following steam generator replacement operating parameters will be closely monitored for consistency with predicted design performance.

7. The proposed Appendix B to DPC-NE-3000-P discusses several custom coding changes made by the RETRAN-3D code developers at the request of Duke Power Company. These include [

]. So that the staff can review these changes and the effects on code results, please provide the source code for the RETRAN-3D subroutines that have been changed. Please provide this information in electronic form.

Response: This information will be mailed to the NRC Project Manager on a diskette.

8. Please describe the modeling assumptions for critical flow from the steam generators following a main steam line break. Page 1-21 of the proposed Appendix C to DPC-NE-3000-P states that the [

] If liquid entrainment out the break is assumed, please justify this assumption. Provide comparisons to assumptions for liquid entrainment from a broken steam line in the previous RETRAN-02 model for Oconee.

Response:

[

]

**Questions 9-27 relate to proposed changes to DPC-NE-3005-PA Revision 1, that will be incorporated to produce new DPC-NE-3005-P Revision 2.**

9. Section 2.2.2 describes the RETRAN-3D Simulation model for Oconee. The section does not discuss input options used in conjunction with the 1979 ANS standard that will be used. Information Notice 96-39 discusses the sensitivity of the ANS standard to various input assumptions to the model. Please discuss how these inputs will be calculated and how they will be made conservative. What corrections will be made to the decay heat predicted by the standard to account for uncertainty?

Response: When decay heat is calculated by RETRAN, a table of multipliers appropriate for the transient is employed to bound the decay heat predicted by an external calculation of decay heat. The external calculation employs the ANSI/ANS-5.1-1979 standard. This calculation uses physics data extracted from representative Oconee core designs as input. Therefore the factors highlighted in IN 96-39, notably the R and Ψ-factors, are derived from design data applicable to Oconee. The Information Notice also highlights three additional assumptions: 1) the capture factor, or G-factor, 2) power history and 3) the number of fissile elements modeled. The calculation uses the standard's G-factor, attributes power to U-235, U-238 and Pu-239 fissile isotopes, and uses a power history representative of 3 batches of fuel burned for up to three cycles. The uncertainty associated with the decay heat prediction, as identified in the standard, is incorporated at a 95% confidence level.

10. Section 5.2.1 of DPC-NE-3005 describes modeling of Oconee for analysis of startup accidents using RETRAN-02. Will RETRAN-3D be used to model startup accidents for Oconee? If so, please provide an equivalent section describing the Oconee model using RETRAN-3D.

Response: Chapter 5 describes the modeling of the startup accident with RETRAN-02 for the original steam generators. The methodology remains the same for the replacement steam generators, with the exception that RETRAN-3D will be used. The following text will be inserted as a new second paragraph in Section 5.1.1 to clarify the transition to RETRAN-3D.

*"The methodology that follows is applicable to the analysis for the original steam generators. The replacement steam generators are analyzed with RETRAN-3D instead of RETRAN-02. No other changes in the methodology that follows are associated with this change in code version or the replacement steam generators. Figure 5-1 is the RETRAN-02 nodalization, and Figure B-2 of Reference 5-3 is the replacement steam generator nodalization."*

11. Section 6.2 of DPC-NE-3005 describes modeling of Oconee for analysis of rod withdrawal at power using RETRAN-02. Will RETRAN-3D be used to model rod withdrawal at power for Oconee? If so, please provide an equivalent section describing the Oconee model using RETRAN-3D.

Response: Chapter 6 describes the modeling of the rod withdrawal at power accident with RETRAN-02 for the original steam generators. The methodology remains the same for the replacement steam generators, with the exception that RETRAN-3D will be used. The following text will be inserted as a new third paragraph in Section 6.0 to clarify the transition to RETRAN-3D.

*"The methodology that follows is applicable to the analysis for the original steam generators. The replacement steam generators are analyzed with RETRAN-3D instead of RETRAN-02. No other changes in the methodology that follows are associated with this change in code version or the replacement steam generators."*

12. In Section 6.5 on page 6-11 the commitment to check the maximum allowable radial peaking limits is to be removed. This is justified from experience showing that rod withdrawal at power is not a limiting transient with regard to the DNBR limit. Please describe the anticipated transient that gives the limiting values for DNBR and compare typical DNBR values calculated for rod withdrawal at power to the limiting values.

Response: The minimum DNBR for the rod withdrawal at power event is 1.959 (BWU-Z CHF correlation) at reference core peaking factors (radial pin peak = 1.714, 1.5 axial peak at X/L = 1.5). Several other UFSAR Chapter 15 events that are not allowed to exceed the DNBR limit have lower DNBR values, including the two-pump coastdown event, dropped rod, small steam line break, and large steam line break. For several of these events the DNBR margin is less than 10%, which means that the core power peaking approaches the maximum allowable peaking (MAP) limits. Therefore, the DNBR for those events approaches 1.40, which is the design DNBR limit including 5% margin for Mk-B11 fuel with the BWU-Z correlation. The MAP limits for the rod withdrawal at power event are never approached since the MAP limits for more limiting events are lower.

13. Section 8.0 of DPC-NE-3005 describes modeling of Oconee for analysis of cold water accidents using RETRAN-02. Will RETRAN-3D be used to model cold water accidents for Oconee? If so, please provide an equivalent section describing the Oconee model using RETRAN-3D.

Response: Chapter 8 describes the modeling of the cold water accident with RETRAN-02 for the original steam generators. The methodology remains the same for the replacement steam generators, with the exception that RETRAN-3D will be used. The following text will be inserted as a new third paragraph in Section 8.0 to clarify the transition to RETRAN-3D.

*"The methodology that follows is applicable to the analysis for the original steam generators. The replacement steam generators are analyzed with RETRAN-3D instead of RETRAN-02. No other changes in the methodology that follows are associated with this change in code version or the replacement steam generators."*

14. Section 9.2 of DPC-NE-3005 describes modeling of Oconee for analysis of loss of flow accidents using RETRAN-02. Will RETRAN-3D be used to model loss of flow accidents for Oconee? If so, please provide an equivalent section describing the Oconee model using RETRAN-3D.

Response: Chapter 9 describes the modeling of the loss of flow accidents with RETRAN-02 for the original steam generators. The methodology remains the same for the replacement steam generators, with the exception that RETRAN-3D will be used. The following text will be inserted as a new second paragraph in Section 9.1.1 to clarify the transition to RETRAN-3D.

*"The methodology that follows is applicable to the analysis for the original steam generators. The replacement steam generators are analyzed with RETRAN-3D instead of RETRAN-02. No other changes in the methodology that follows are associated with this change in code version or the replacement steam generators."*

15. Section 10.2.1 of DPC-NE-3005 describes modeling of Oconee for analysis of locked rotor accidents using RETRAN-02. Will RETRAN-3D be used to model locked rotor accidents for Oconee? If so, please provide an equivalent section describing the Oconee model using RETRAN-3D. Please discuss the conservatism of using an equilibrium pressurizer model to predict the peak reactor system pressure following locked rotor analysis in comparison to the non-equilibrium pressurizer option in RETRAN-3D.

Response: Chapter 10 describes the modeling of the locked rotor accident with RETRAN-02 for the original steam generators. The methodology remains the same for the replacement steam generators, with the exception that RETRAN-3D will be used. The following text will be inserted as a new third paragraph in Section 10.1.1 to clarify the transition to RETRAN-3D.

*"The methodology that follows is applicable to the analysis for the original steam generators. The replacement steam generators are analyzed with RETRAN-3D instead of RETRAN-02. No other changes in the methodology that follows are associated with this change in code version or the replacement steam generators."*

The non-equilibrium pressurizer model is used in the locked rotor analysis. Refer to the response to Question #2.

16. Section 11.1 of DPC-NE-3005 describes modeling of Oconee for analysis of control rod misalignment accidents using RETRAN-02. Will RETRAN-3D be used to model control rod misalignment accidents for Oconee? If so, please provide an equivalent section describing the Oconee model using RETRAN-3D.

Response: Chapter 11 describes the modeling of the control rod misalignment accident with RETRAN-02 for the original steam generators. The methodology remains the same for the replacement steam generators, with the exceptions that RETRAN-3D will be used, and that a revised Section 11.1.5 discussion of excore flux instrumentation has been included. The following text will be inserted as a new fourth paragraph in Section 11.1 to clarify the transition to RETRAN-3D.

*"The methodology that follows is applicable to the analysis for the original steam generators. The replacement steam generators are analyzed with RETRAN-3D instead of RETRAN-02. No other changes in the methodology that follows are associated with this change in code version or the replacement steam generators."*

17. The additions to Section 11.1.5 and Section 16.1.3 describe new methodology in calculating the effect of a reduction in reactor vessel downcomer water temperature on the excore flux detector. The new methodology uses the SAS2H/ORIGEN-S models of the SCALE code system and the MCNP Monte-Carlo N-particle transport code. The current methodology is described as a synthesis of plant data and analysis results obtained from Framatome Advanced Nuclear Products. Please provide a comparison of the analytical results from both methods for a control rod misalignment accident and for a small steam line break accident and discuss the relative accuracy of each method.

Response: The MCNP models include all significant elements of the physical geometry, materials and neutron energy spectrum, and thus include the necessary detail to ensure that all problem-significant phase-space has been sampled. Robust Monte Carlo variance reduction for this problem facilitates accelerated convergence, thus yielding very small relative errors and achieving highly accurate results. The MCNP computer code is well-suited for this analysis, and the models used for this calculation are similar in nature to typical vessel fluence calculations. The viability and accuracy of MCNP for vessel and ex-vessel neutron transport calculations has been previously demonstrated (John. C. Wagner, "Monte Carlo Transport Calculations and Analysis for Reactor Pressure Vessel Neutron Fluence," M.S. Thesis, The Pennsylvania State University, College of Engineering, December 1994. See also Wagner, Haghighat, Petrovic, "Monte Carlo Transport Calculations and Analysis for Reactor Pressure Vessel Neutron Fluence," Nuclear Technology, Vol. 114, June 1996). Additionally, the results of quarter-core models constructed for this problem compare favorably with empirical plant data (i.e., from $T_{AVG}$ reduction evolutions during end-of-cycle operations).

18. Describe additional details of how the SAS2H/ORIGEN-S/MCNP methodology is utilized to calculate the effect of reduction in reactor vessel downcomer water temperature on the excore flux detector signal. Describe the input for each code and how the output is utilized. Describe how the source distribution is calculated for the core for input into the MCNP code to evaluate misaligned control rods and main steam line breaks.

Response: Detailed input description for the SAS2H/ORIGEN-S modules of the SCALE Code System and to the MCNP Code System is provided in the references supplied in the response to question 19 below. The SAS2H/ORIGEN-S calculation is performed to characterize irradiated fuel. The SAS2H analyses use fuel assembly pin-cell dimensions and materials to develop problem-dependent cross section sets (using the XSDRNPM, BONAMI and NITAWL modules of SCALE) for fuel burnup and depletion calculations performed with ORIGEN-S. The constituent actinides and fission products resulting from SAS2H/ORIGEN-S depletion calculations are used as input to the fuel material description in the MCNP model.

In the MCNP model, the fuel assemblies are modeled as a repeating lattice of fuel pins, guide tubes and instrument tubes. This lattice is filled into the fuel region inside the core baffle plates. The core baffle plate and former plate materials and dimensions are modeled explicitly. A set of concentric cylinders is used to model the core barrel, thermal shield, downcomer region, reactor vessel, and concrete shield wall (to model neutron thermalization in the concrete and backscatter to the detector). The MCNP input is built as a quarter-core geometry model. The result of the transport calculation at the excore detector location is determined by a tally on $B^{10}$ absorptions (the detectors are uncompensated ion chambers that function by the following reaction: $B^{10}[n,\alpha]Li^{7}$).

The Monte Carlo tally results for the detector location are compared for moderator density statepoints in order to obtain comparative results for detector response versus a known temperature change. This result is used as input to safety analysis calculations as a simple neutron attenuation coefficient (percent loss of indicated power per degree moderator temperature change over the applicable temperature range). Various sensitivity studies have been completed, including, but not limited to, (a) an assessment of sensitivity of detector response to source distribution (i.e., radial power shape), and (b) an assessment of detector response to neutron energy spectrum assumptions. While the absolute detector response is affected by changes in radial source distribution, the relative detector response due to moderator density change is unaffected (using the same radial power shape for each moderator density statepoint). Therefore, the fractional decrease in existing detector response is governed by moderator density rather than the radial power shape. The relative detector response to different moderator densities was also compared for two neutron energy spectrum models. In the first model, the problem was evaluated as a watt-fission spectrum in a fission system (i.e., with fissions allowed in the fuel). In the second model, the problem was evaluated as a fixed-neutron energy shielding and transport problem, where neutron energy bin probabilities were obtained from the CASMO computer code and used as input to MCNP, and fissions were turned off in the fuel. The relative detector response to changes in moderator density was unaffected by the selection of source models. In further demonstration of the adequacy of source modeling assumptions, a tally was performed (on fuel elements) to compare the two source models discussed above, and the neutron energy bin probabilities for the built-in MCNP watt-fission spectrum with fissions compares favorably with the CASMO neutron energy spectrum without fissions.

19. Please provide dates for new references 11-5, 11-6, and 11-7 which are given again as new references 16-4, 16-5, and 16-6.

Response: References 11-5 and 11-6 are dated March 2000, and refer to version 4.4 of the SCALE Code System. Reference 11-7 is dated March 2000, and refers to version 4C of the MCNP Code System. As code errors are corrected and new code updates are issued by the code authors, these updates will be implemented under internal Duke Energy software procedures.

20. Section 12.2 of DPC-NE-3005 describes modeling of Oconee for analysis of turbine trip using RETRAN-02. Will RETRAN-3D be used to model turbine trip for Oconee? If so, please provide an equivalent section describing the Oconee model using RETRAN-3D. Please discuss the conservatism of using an equilibrium pressurizer model to predict the peak reactor system pressure following turbine trip in comparison to the non-equilibrium pressurizer option in RETRAN-3D. Provide new FSAR text and figures (this was requested during a telecon on October 3, 2002)

Response: Chapter 12 describes the modeling of the turbine trip with RETRAN-02 for the original steam generators. The methodology remains the same for the replacement steam generators, with the exception that RETRAN-3D will be used. The following text will be inserted as a new second paragraph in Section 12.1 to clarify the transition to RETRAN-3D.

*"The methodology that follows is applicable to the analysis for the original steam generators. The replacement steam generators are analyzed with RETRAN-3D instead of RETRAN-02. No other changes in the methodology that follows are associated with this change in code version or the replacement steam generators."*

The non-equilibrium pressurizer model is used in the turbine trip analysis. Refer to the response to Question #2. The revisions to the UFSAR text and associated tables and figures are included at the end of Attachment 1.

21. Section 13.1 of DPC-NE-3005 describes modeling of Oconee for analysis of steam generator tube rupture using RETRAN-02. Will RETRAN-3D be used to model steam generator tube rupture for Oconee? If so, please provide an equivalent section describing the Oconee model using RETRAN-3D. Will the special modeling provisions to [ ] that was used for the RETRAN-02 model be used for the RETRAN-3D analysis? How will the [ ] that was added to RETRAN-3D for DPC be applied to steam generator tube rupture analysis?

Response: Chapter 13 describes the modeling of the steam generator tube rupture accident with RETRAN-02 for the original steam generators. The methodology remains the same for the replacement steam generators, with the exception that RETRAN-3D will be used. The following text will be inserted as a new third paragraph in Section 13.0 to clarify the transition to RETRAN-3D.

*"The methodology that follows is applicable to the analysis for the original steam generators. The replacement steam generators are analyzed with RETRAN-3D instead of RETRAN-02. No other changes in the methodology that follows are associated with this change in code version or the replacement steam generators."*

[ ] In the steam generator tube rupture analysis the RCS cooldown rate is being manually controlled. Therefore, there is no need for special modeling to ensure a conservative prediction of the cooldown rate.

22. Section 14.1.3 of DPC-NE-3005 describes modeling of Oconee for analysis of the peak pressure from a rod ejection accident using RETRAN-02. Will RETRAN-3D be used to model rod ejection accidents for Oconee? If so, please provide an equivalent section describing the Oconee model using RETRAN-3D. Please discuss the conservatism of using an equilibrium pressurizer model to predict the peak reactor system pressure following a rod ejection accident in comparison to the non-equilibrium pressurizer option in RETRAN-3D.

Response: Chapter 14 describes the modeling of the rod ejection accident with RETRAN-02 for the original steam generators. The methodology remains the same for the replacement steam generators, with the exception that RETRAN-3D will be used. The following text will be inserted as a new second paragraph in Section 14.1.1 to clarify the transition to RETRAN-3D.

*"The methodology that follows is applicable to the analysis for the original steam generators. The replacement steam generators are analyzed with RETRAN-3D instead of RETRAN-02. No other changes in the methodology that follows are associated with this change in code version or the replacement steam generators."*

The non-equilibrium pressurizer model is used in the rod ejection analysis. Refer to the response to Question #2.

23. Section 15.2.1.3 describes the steam generator model to be used for main steam line break analysis. The proposed modifications for Revision 2 add a lead-in paragraph saying the discussion applies only to the existing steam generators analyzed with RETRAN-02. Please provide a revised Chapter 15 for DPC-NE-3005-PA that describes the RETRAN-3D analytical model of main steam line break with the new steam generators in service.

Response: Chapter 15 describes the modeling of the steam line break accident with RETRAN-02 for the original steam generators. Most of the methodology remains the same for the replacement steam generators, with the exception that RETRAN-3D will be used. Refer to Revisions 39, 40, and 41 for specific changes. Also refer to Attachment 5 of this submittal for new revisions (Items 3-6) that have not been previously submitted for NRC review. The following revisions will be implemented to clarify the differences.

The following text will be inserted as a new second paragraph in Section 15.1.1 to clarify the transition to RETRAN-3D.

*"The methodology that follows, unless specified otherwise, is applicable to both the analysis of the original steam generators with RETRAN-02 and the analysis of the replacement steam generators with RETRAN-3D.. The following changes in the methodology are specific to RETRAN-3D:*

- *Steam generator modeling differences in Section 15.2.1.5*
- *Steam generator water carryout control differences in Section 15.2.1.6*
- *Reactor coolant pump modeling differences in Section 15.3.1.1.2"*

The title of Section 15.2.1.4 will be revised to "Original Steam Generator RETRAN-02 Model". The proposed Revision #38 that added a lead in paragraph will be withdrawn.

The following new Section 15.2.1.5, "Replacement Steam Generator RETRAN-3D Model", will be inserted to describe all steam line break RETRAN-3D and ROTSG modeling differences relative to the RETRAN-3D base model. Existing Sections 15.2.1.5 and 15.2.1.6 will be resequenced to 15.2.1.6 and 15.2.1.7.

*"15.2.1.5 Replacement Steam Generator RETRAN-3D Model*

*Zero steam generator tube plugging is assumed to maximize the primary-to-secondary heat transfer. The choking option (Extended Henry and Moody) is turned on at the steam outlet nozzle to account for the flow restricting orifice. The vertical junction option is used and the inertia is increased for the aspirator junctions to smooth the enthalpy and mass flow rate predictions. The isoenthalpic expansion choked flow option is used at the aspirator junctions to avoid junction enthalpy errors should the enthalpy decrease to below 170 Btu/lbm, which is the Moody model limit. The junction flow area for the steam generator cold leg outlet nozzle in the unaffected loop is increased if a code abort occurs."*

24. Page 5-36 to the revisions to DPC-NE-3005-PA states that the only significant difference between the transient thermal-hydraulic behavior with the new steam generators will be for the main steam line break analysis, in which the flow restriction orifices in the outlet nozzles will effectively reduce the minimum break size and blowdown rate. In addition the new steam generators will have a greater heat transfer area and a greater water mass. Differences in calculational models between RETRAN- 02 and RETRAN-3D may also affect the result. In particular assumptions for phase separation within the steam generators and steam flow rate from the break may affect the result. Please provide main steam line break analyses similar to those in Chapter 15 of DPC-NE-3005-P for cases both with and without offsite power for the new steam generators with RETRAN-3D.

Response: The revised topical report text and figures (included after the response to Question 29) show the RETRAN-3D main steam line break analysis results for both with and without offsite power with the replacement steam generators. A brief summary of the result of the analyses is as follows. The steam outlet nozzle flow restrictors cause a slower RCS cooldown and depressurization rate, which provide more favorable results. For the with offsite power case the slower cooldown rate does not result in a loss of the subcritical margin according to the RETRAN-3D and SIMULATE simulations. Consequently, no detailed core thermal-hydraulic analysis was performed. For the without offsite power case, the RCS depressurization rate is slower, and the statepoint at the time of minimum DNBR is more favorable. Consequently an increase in DNBR margin is gained relative to the previous RETRAN-02 OTSG analyses.

25. Section 15.2.1.5 describes models used to minimize steam line moisture carryout of a broken steam line. Will these models be implemented in the RETRAN-3D analyses with the new steam generators? Please discuss the relationship between the modeling assumptions of Section 15.2.1.5 and the custom modifications to RETRAN-3D that [ ] in the RETRAN-3D analyses will be conservative for predicting reactor system overcooling.

Response: The custom modifications to RETRAN-3D that [ ] are only used in the mass and energy release analysis in topical report DPC-NE-3003-P. They are not used for the Chapter 15 core response steam line break analysis in either the original methodology using RETRAN-02, or the revised methodology using RETRAN-3D. There are some differences in the modeling approach which are being submitted for NRC review in Attachment 5, Item 4. The differences in the RETRAN-3D modeling approach are as follows:

26. Section 15.3.1.1 discusses the RETRAN-02 analysis for the offsite power available case. The subsection describing reactor coolant pump modeling states that the RCPs in the unaffected loop are tripped at 100 seconds to avoid a code error associated with pressure oscillations in this loop. Does the same code error exist in RETRAN-3D? Will the reactor coolant pumps be assumed to trip at 100 seconds as in the RETRAN-02 analysis?

Response: In the RETRAN-3D methodology it is no longer necessary to trip the reactor coolant pumps in the unaffected loop to avoid a code error at 100 seconds. The pumps remain in operation for the with offsite power case. Item 5 of Attachment 5 requests NRC review of this change to the methodology. One of the main reasons for this change is a revision to the reactor coolant pump two-phase degradation model, which has been revised in the Oconee and the McGuire/Catawba RETRAN base models. The new model, which is based on the CE/EPRI test data results in less head degradation during the steam line break analysis. The Oconee RETRAN-3D model with this revised pump degradation modeling is able to run through the steam line break analysis with voiding in the unaffected loop without aborting. This revised reactor coolant pump degradation model is submitted for NRC review in Attachment 3 to this submittal.

27. Section 15.3.1 under Main Feedwater System states that the limiting assumption with respect to maximizing the overcooling and reactivity addition has been determined by analysis to be the case with the ICS controlling MFW to the minimum steam generator level setpoint including uncertainty. Justify that this statement is still true with the new steam generators using RETRAN-3D.

Response: Analyses using RETRAN-3D for the replacement steam generators have been performed for the case with uncontrolled main feedwater, which leads to overfilling the steam generator, and with the ICS controlling the level to a setpoint. The results of the analyses show that the case with uncontrolled main feedwater flow is slightly worse than the case with the ICS controlling the steam generator level. For both cases there is a subcritical margin, so neither case approaches the DNBR limit or any other limit. This significant change in the results, compared to the RETRAN-02 analyses for the original steam generators is due to the flow restricting steam outlet nozzles on the replacement steam generators. These nozzles effectively limit the break size, which limits the cooldown rate and the rate of positive reactivity addition. This slowing down of the progression of the event allows more time for the boric acid from the HPI System and the core flood tanks to be delivered to limit the total reactivity to a negative value. These new results indicate that the main feedwater boundary condition, although an important boundary condition, produces essentially the same overall results.

28. To enable the NRC staff to perform audit calculation and sensitivity analyses if needed, please provide an electronic copy of the RETRAN-3D input deck used to perform main steam line break analysis.

Response: The NRC Project Manager will be provided with a diskette with the requested input deck.

29. Section 16.1 of DPC-NE-3005 describes modeling of Oconee for analysis of small steam line breaks using RETRAN-02. Will RETRAN-3D be used to model small steam line breaks for Oconee? If so, please provide an equivalent section describing the Oconee model using RETRAN-3D. Provide new FSAR text and figures (this was requested during a telecon on October 3, 2002)

Response: Chapter 16 describes the modeling of the small steam line break accident with RETRAN-02 for the original steam generators. Most of the methodology remains the same for the replacement steam generators, with the exception that RETRAN-3D will be used. Refer to Revisions 43 and 44 for other specific changes. Also refer to Attachment 5 of this submittal for new revisions (Items 10-12) that have not been previously submitted for NRC review. The following text will be inserted as a new third paragraph in Section 16.1.1 to clarify the transition to RETRAN-3D.

*"The methodology that follows, unless noted otherwise, is applicable to the analysis for the original steam generators. The replacement steam generators are analyzed with RETRAN-3D instead of RETRAN-02. The modeling of the aspirator port in the steam generator secondary has been revised in the ROTSG methodology. A RETRAN control system is used to stop steam flow through the aspirator port when the downcomer floods, since this is judged to be non-physical. Also, the methodology for selection of physics parameter values has been changed to maintain consistency with the time in core life. No other changes in the methodology that follows are associated with this change in code version or the replacement steam generators."*

The revisions to the UFSAR text and associated tables and figures are included at the end of Attachment 1.

**The Following is Supplemental Information Related to Question #24**

**Revised DPC-NE-3005-P Chapter 15 - Steam Line Break**

Note: The revised content is in italics. Deleted content is not shown.

# 15.0 STEAM LINE BREAK

## 15.1 Overview

### 15.1.1 Description

The steam line break accident initiates with a double-ended rupture of one of the two main steam lines. Since the two steam lines are connected in the steam chest between the turbine stop valves and the control valves, the break initially results in a rapid blowdown of both steam generators. The steam generator depressurization initiates a rapid Reactor Coolant System (RCS) cooldown leading to a reactor trip on low RCS pressure or variable low RCS pressure within the first few seconds of the accident. The reactor trip causes the turbine stop valves to close, isolating the affected steam generator from the unaffected steam generator. Main feedwater flow to each steam generator will be controlled by the Integrated Control System (ICS) by maintaining a minimum post trip steam generator level. If main feedwater is available and controlling steam generator level to the ICS setpoint, emergency feedwater will not be actuated. If main feedwater is lost, or if the ICS fails to control feedwater flow to the affected steam generator, emergency feedwater is likely to be actuated. The affected steam generator continues to depressurize, while the pressure in the isolated steam generator repressurizes and is controlled by the turbine bypass valves and possibly the main steam safety valves. Auxiliary steam loads may also depressurize the isolated steam generator. The cooldown of the RCS continues, resulting in reverse heat transfer in the isolated steam generator. The cooldown of the RCS caused by the continued addition of main and/or emergency feedwater to the depressurized steam generator may lead to a loss of shutdown margin and a return-to-power. Any return-to-power is eventually shut down by the boron injected from the High Pressure Injection (HPI) System and core flood tanks (CFTs).

*The methodology that follows, unless specified otherwise, is applicable to both the analysis of the original steam generators with RETRAN-02 and the analysis of the replacement steam* *generators with RETRAN-3D.. The following changes in the methodology are specific to RETRAN-3D:*

- *Steam generator modeling differences in Section 15.2.1.5*
- *Steam generator water carryout control differences in Section 15.2.1.6*
- *Reactor coolant pump modeling differences in Section 15.3.1.1.2*

### 15.1.2 Acceptance Criteria

The acceptance criteria for the steam line break accident are as follows:

- The core will remain intact for effective core cooling, assuming minimum tripped rod worth with a stuck rod.
- Doses will be within 100% of 10CFR100 limits.

The steam line break analysis is performed assuming a stuck control rod, a single failure in the Engineered Safety Features or the Emergency Feedwater System, and with consideration of both offsite power maintained and offsite power lost. Fuel failure will be assumed for any fuel pin that exceeds the DNBR limit.

### 15.1.3 Analytical Approach

The steam line break transient requires a limiting set of physics parameters to be determined for use as initial and boundary conditions. These parameters are input to the Oconee RETRAN-02 model (References 15-1 and 15-2) for the system thermal-hydraulic analysis. The with offsite power RETRAN-02 analysis generates the transient core thermal-hydraulic boundary conditions (core heat flux, core inlet flow, core inlet temperature and core exit pressure). The steam line break with offsite power is a severe overcooling transient which results in a return-to-power condition. The affected loop cold leg temperatures are much colder than the unaffected loop and cause asymmetric core inlet temperature conditions. To simulate this asymmetric condition properly, a[ ]channel VIPRE-01 (Reference 15-3) model is used (Figure 15-1). A statepoint DNBR calculation is performed since the return-to-power during the steam line break accident is slow and a statepoint analysis provides conservative DNBR results. The RETRAN-02 thermal-hydraulic statepoint is analyzed using the SIMULATE-3P (Reference 15-4) code to determine a detailed core power distribution including a stuck rod. The detailed core power distribution and the statepoint conditions are then analyzed with the VIPRE-01 code to determine the minimum DNBR.

For the without offsite power case, the core thermal-hydraulic boundary conditions from the RETRAN-02 analysis are input to the Oconee VIPRE-01[ ]channel model (Reference 15-2) to determine the DNBR statepoint. The VIPRE-01 model is then utilized to calculate a set of maximum allowable radial peaking (MARP) limits such that DNB will not occur. The MARP limits are compared against the SIMULATE-3P core power distribution to determine the number of fuel pins exceeding the DNB limit and therefore assumed to fail.

## 15.2 Simulation Codes and Models

### 15.2.1 RETRAN-02

The RETRAN-02 Oconee base model described in Section 2.2.1 of Reference 15-2 is utilized for the steam line break analysis except as described below. The steam line break model has been previously submitted (Reference 15-5) and approved by the NRC for the Oconee steam line break accident mass and energy release modeling.

#### 15.2.1.1 Nodalization of Reactor Vessel

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15.2.1.2 Transport Delay Model

*The transport delay model is known to produce anomalous predictions if flow reversals occur. This model is turned off in the primary loop piping volumes if flow reversals are predicted in those volumes.*

15.2.1.3 Condensate/Feedwater System Model

A Condensate/Feedwater System model is added to the RETRAN base deck to accurately predict the feedwater flow boundary condition during the steam line break accident. The Condensate/Feedwater System model contains fill tables to simulate the condensate booster pumps and the D heater drain pumps. Homologous pump curves are included to accurately model the main feedwater pumps. Non-conducting heat exchangers are used to model all of the feedwater heaters.

#### *15.2.1.4 Original Steam Generator RETRAN-02 Model*

Low steam generator tube plugging will maximize the transient primary-to-secondary heat transfer. The assumption of low steam generator tube plugging also maximizes the RCS volume, which slightly increases the overall heat capacity of the RCS. Sensitivity studies have been performed and have determined that the impact of the tube plugging on the heat transfer area is the dominant effect. Based upon plant data, a lower bound of 1% tube plugging is modeled.

The vertical junction option is used for the aspirator junctions to smooth the enthalpy and mass flow rate predictions through the aspirator port during the accident. This is necessary due to the reverse flow predicted through these junctions during the accident. The inertia for these junctions is also increased in order to minimize the rate of change in flow through the aspirator ports. The choking option (Extended Henry and Moody) is turned on at the steam generator exit nozzle junction, which is reasonable given the high steam velocities. The choking option is turned off at the feedwater nozzle junction, which will result in more feedwater entering the faulted steam generator. The isoenthalpic expansion choked flow option is utilized for Junctions 126, 134, 225, 226 and 234. This avoids junction enthalpy errors when the enthalpy decreases below 170 Btu/lbm (Moody limit). In addition, dynamic slip is modeled in Junctions 136 and 137. This is done in an attempt to minimize the liquid carried into the steam line.

#### *15.2.1.5 Replacement Steam Generator RETRAN-3D Model*

*Zero steam generator tube plugging is assumed to maximize the primary-to-secondary heat transfer. The choking option (Extended Henry and Moody) is turned on at the steam outlet nozzle to account for the flow restricting orifice. The vertical junction option is used and the inertia is increased for the aspirator junctions to smooth the enthalpy and mass flow rate predictions. The isoenthalpic expansion choked flow option is used at the aspirator junctions to avoid junction enthalpy errors should the enthalpy decrease to below 170 Btu/lbm, which is the Moody model limit. The junction flow area for the steam generator cold leg outlet nozzle in the unaffected loop is increased if a code abort occurs.*

*15.2.1.6* <u>*Steam Generator Water Carryout Control*</u>

Water carryout during blowdown of the affected steam generator can have the effect of reducing the rate of overcooling, since water that does not boil in the tube bundle region will not absorb the heat of vaporization. A secondary concern with water carryout in a steam line break analysis is that the break flow with two-phase conditions will be considerably less on a volumetric basis than single-phase steam flow, and will thus slow the rate of steam generator depressurization. This in turn slows the decrease in steam generator saturation temperature and the primary-to-secondary heat transfer rate, which is non-conservative. However, with uncontrolled main and/or emergency feedwater flow, steam generator overfill will eventually occur. Water carryout at that time is realistic. Possible unrealistic or non-conservative water carryout is addressed in the model.

#### 15.2.1.7 Break Model

The break is modeled by dividing the ruptured main steam line into two volumes with a connecting junction, and by adding the two break junctions. The full cross-sectional area of the 34" main steam line is 6.3 ft$^2$. Thus, the double-ended break of the 34" main steam line results in a total initial break flow area of 12.6 ft$^2$.

*The replacement steam generators have flow restricting orifices in the steam outlet nozzles. These reduce the critical flow area to 1.804 ft$^2$ per steam generator, and decrease the blowdown rate relative to the original steam generators.*

### 15.2.2 VIPRE-01

The VIPRE-01 code is used for the steam line break core thermal-hydraulic analyses. VIPRE-01 thermal-hydraulic boundary conditions (core exit pressure, core inlet temperature, core inlet flow, and heat flux) are obtained from the RETRAN-02 system transient simulation. Since the

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moderator heating, flow correlations, and other correlations is identical to that described in Reference 15-2. The subcooled and bulk void correlations are different than those in Reference 15-2, and are described in Section 15.3.1.2.3. The critical heat flux (CHF) correlations used to evaluate the DNBR are the Westinghouse W-3S (Reference 15-3, Appendix D) for the Mk-B10 or Mk-B11 fuel types, and the BWU (References 15-7 and 15-8) correlations for Mark-B11 fuel.

For the without offsite power analysis, the [ ] channel VIPRE-01 model described in Reference 15-2 is used to calculate the transient local coolant properties and DNBR. The BWC (Reference 15-6) and BWU CHF correlations are used to perform the DNBR calculations for the Mk-B10T and Mk-B11 fuel assembly types, respectively. The VIPRE-01 analysis employs the SCD methodology for the offsite power lost case.

### 15.2.3 SIMULATE-3P

SIMULATE-3P is used to generate safety analysis physics parameters and three-dimensional core pin power distributions. The system transient response during a steam line break accident is sensitive to core temperature feedback. The moderator reactivity versus temperature and the Doppler reactivity versus fuel temperature curves are selected such that the most limiting conditions, which occur at end-of-cycle (EOC), are predicted.

The asymmetric conditions for the with offsite power analysis require non-uniform core inlet temperatures to be input to SIMULATE-3P. The maximum worth stuck rod is conservatively assumed to be in the cold half of the core which will increase the local reactivity and power. A

10% reduction in the worth of the remaining control rods is also assumed. These assumptions result in a conservative reactivity calculation and power distribution at the limiting RETRAN statepoint. The SIMULATE-3P reactivity prediction is used to verify that the RETRAN kinetics model is conservative. The SIMULATE-3P pin power distribution at the limiting RETRAN statepoint is then input to VIPRE for the DNBR analysis.

For the without offsite power analysis, the stuck rod is conservatively assumed to be in the colder half of the core since this will increase the local reactivity and power. SIMULATE-3P is used to calculate the pin power distribution which is used to compare to the MARP limits generated in the VIPRE analysis.

## 15.3 Transient Analysis

The steam line break analysis presented herein is divided into two sections. The first section assumes that offsite power is available, and is concerned with the potential for a post-trip return-to-power and DNB. The second section assumes that offsite power is lost coincident with the opening of the break, and is concerned with the flow coastdown and primary system depressurization effects on DNB.

### 15.3.1 With Offsite Power

#### 15.3.1.1 RETRAN-02 Analysis

##### 15.3.1.1.1 Initial Conditions

The initial conditions for the steam line break analysis with offsite power are selected to maximize the RCS cooldown and depressurization, and thereby maximize the potential for a post-trip return-to-power and DNB. Since the SCD methodology does not cover the range of RCS pressures expected for the cases that assume offsite power is available, a deterministic approach will be utilized in the selection of the initial conditions.

Power Level

Full rated power plus uncertainty is assumed. High initial power level maximizes the initial steam generator inventory and feedwater flow rate, both of which will maximize the primary-to-secondary heat transfer once the break occurs. A steam line break accident from hot zero power (HZP) is not analyzed. At HZP, feedwater is aligned through the startup feedwater control valves, which results in a much lower feedwater flow rate than at full power. Sensitivity studies have been performed and have determined that the steam line break from HZP is bounded.

RCS Pressure

A low initial pressure minimizes the time to reactor trip. An earlier trip reduces the integrated energy deposition into the RCS, leading to lower RCS temperatures. A lower initial pressure is also conservative with respect to DNB. However, a lower initial pressure results in an earlier actuation of the Engineered Safeguards Systems (HPI and CFTs) which inject boron into the RCS and shut down the reactor if a return-to-power occurs. Sensitivity studies have been performed and have determined that a low initial RCS pressure is the most limiting assumption.

Pressurizer Level

A low initial pressurizer level minimizes the volume of relatively hot water that drains into the RCS upon pressurizer outsurge, thereby maximizing the RCS cooldown and any return-to-power. Thus, nominal pressurizer level less uncertainty is assumed.

RCS Temperature

The ICS controls the average coolant temperature at a constant value whenever power is greater than 15%. For the steam line break accident, a lower initial average coolant temperature will result in a greater cooldown of the primary system. This will result in more positive reactivity addition due to the negative moderator temperature coefficient, and thus maximize any return-to-power. Thus, nominal RCS average temperature less uncertainty is assumed.

RCS Flow

Since this transient is being evaluated for minimum DNBR, a low initial RCS flow is used.

Core Bypass Flow

High core bypass flow is assumed which minimizes core flow and is conservative for DNB.

Fuel Temperature

A low initial fuel temperature is used to minimize the stored energy in the fuel. A conservatively low EOC fuel temperature is assumed.

Steam Generator Mass

A conservatively high steam generator mass is assumed to maximize the overcooling.

15.3.1.1.2 Boundary Conditions

The key boundary conditions for the steam line break with offsite power are as follows:

Break Opening Time

A break opening time of 0.1 seconds is assumed. Based on sensitivity studies performed, shorter break opening times do not significantly alter the initial secondary side depressurization.

Reactor Coolant Pump Modeling

The RETRAN two-phase flow degradation model is used for the RCPs since significant voiding is predicted in the unaffected loop. *In the RETRAN-02 analysis for the original steam generators, the RCPs in the unaffected loop are tripped at 100 seconds to avoid a code error associated with pressure oscillations in this loop due to two-phase performance. Tripping the RCPs in the unaffected loop has a conservative impact on the cooldown of the RCS since there is reverse heat transfer taking place in the steam generator." In the RETRAN-3D analysis for the replacement steam generators this code error does not occur and the RCPs are not tripped*

Turbine Stop Valves

A slow turbine stop valve stroke time (1.0 second) is assumed to isolate the unaffected steam generator from the affected steam generator. This maximizes the overcooling.

Main Steam Safety Valves

The main steam safety valves are modeled using conservative assumptions for drift, blowdown and valve capacity that maximize relief flow and minimize the secondary pressure response in the unaffected steam generator. A lower pressure will minimize the reverse primary-to-secondary heat transfer in this steam generator, and maximize the RCS cooldown.

Extraction Steam

To maximize the cooldown of the RCS, it is conservative to model the steam loads on the isolated steam generator. A conservatively high extraction steam flow rate is assumed.

Decay Heat

To maximize the RCS cooldown, a low decay heat power level assuming a multiplier of 0.9 is applied to the 1979 ANS Standard 5.1 decay heat power.

Single Failure

The analysis examines a single failure of the EFW control valve to the affected steam generator or a single failure of the Engineered Safeguards that results in only one train of HPI.

15.3.1.1.3 Physics Parameters

Moderator Temperature Feedback

A table of reactivity as a function of moderator density is input to account for moderator reactivity effects. The consequences of a steam line break accident are more severe at EOC due to the more negative moderator temperature coefficient. The most negative EOC moderator temperature feedback curve is used in the analysis.

Doppler Temperature Feedback

A table of reactivity as a function of fuel temperature is input to model Doppler reactivity effects. The most negative Doppler curve is used in the analysis. However, the most negative Doppler curve will also result in the largest negative feedback during any return-to-power. Since the boron injected by the HPI system and CFTs limits the return-to-power (rather than the negative reactivity due to Doppler feedback) it is conservative to assume a most negative Doppler curve.

Reactivity Weighting

[ ]

Beta-effective and Neutron Lifetime

A small value of $\beta_{eff}$ and prompt neutron lifetime are chosen to maximize the power decrease on reactor trip. Small values of these parameters will also enhance any return-to-power. EOC decay constants and delayed neutron precursor fractions are also assumed.

Scram Curve and Worth

The control rods are inserted when the reactor trips. For this analysis, a top-peaked scram curve and a lower bound on the rod insertion time are assumed. These assumptions minimize the post-trip energy addition to the RCS, leading to a greater cooldown. A scram worth is selected which maintains a reactivity margin between the RETRAN-02 and SIMULATE-3P reactivity predictions at the limiting RETRAN statepoint.

Boron Reactivity

Differential Boron Worth

A differential boron worth is used to model the reactivity addition from the boron injected by the HPI pumps and the CFTs. A low differential boron worth (% $\Delta k/k$/ppm) is conservative in that it will minimize the negative reactivity added by these systems.

15.3.1.1.4 Control, Protection, and Safeguards Systems

Reactor Control

Following the steam line break, the combined effect of decreasing turbine header pressure and T-ave would result in an increase in reactor demand to the high limit. Since a reactor trip will occur within the first few seconds of the accident, it is reasonable to make the simplifying assumption that the control rods are in manual control.

Reactor Trip

An early reactor trip is conservative in that it minimizes the integrated energy transferred into the RCS, leading to a more severe cooldown. Thus, the variable low pressure trip and low RCS pressure trip setpoints are adjusted to ensure an early reactor trip occurs. A lower bound on the delay time for both trip functions is used.

RCS Pressure Control

No credit is taken for pressurizer heater operation. Due to the rapid depressurization of the RCS, the pressurizer sprays, PORV, and safety valves are not actuated.

Pressurizer Level Control

No credit is taken for the automatic operation of makeup and letdown to attempt to maintain pressurizer level. The makeup and letdown flows are assumed to isolate simultaneously and to be balanced prior to isolation. Not taking credit for the makeup and letdown is conservative for the evaluation of minimum DNBR.

Emergency Core Cooling System

Minimum HPI flow is conservative for the steam line break, since the injected borated water from this system helps to prevent or terminate any return-to-power as well as repressurize the RCS. The HPI system is simulated using fill tables that model the A and B HPI pumps injecting through the A train and the C HPI pump injecting through the B train. Sensitivity studies have examined the effect of a failure in the 4160V switchgear or the failure of the EFW control valve to the affected steam generator. For the cases that assume an EFW control valve single failure, three HPI pump minimum flow is credited. Since reverse flow is established in the unaffected loop, the A and B pump flow is injected in the unaffected loop. This maximizes the flowpath the injected boron must take to reach the core inlet, which delays the boron negative reactivity addition. For cases that assume the failure of one of the three available 4160V switchgear, the A train of HPI is assumed to be lost. This results in the C pump injecting through the B train for the first 10 minutes. The C pump injected flow occurs in the unaffected loop to maximize the delay in the boron negative reactivity addition. The presence of any unborated water initially in the HPI piping is modeled. A conservative minimum boron concentration is also assumed.

Similarly, the boron concentration in the CFTs is assumed to be a conservative minimum value. Lower bounds on the initial CFT inventory, pressure and temperature are also assumed. These assumptions will delay CFT injection, minimize the available inventory of borated water and maximize the RCS cooldown.

Main Feedwater System

Since a reactor trip occurs within the first few seconds of the accident, changes in feedwater control over this period of time will have a negligible impact on the accident. Following reactor trip, the ICS rapidly decreases feedwater demand to zero, and then feedwater flow is restored when steam generator level drops below the minimum level control setpoint. With the ICS in manual MFW flow will continue and, assuming no credit for ICS control or operator action, steam generator overfill will occur. The limiting assumption with respect to maximizing the overcooling and reactivity addition has been determined by analysis to be the case with the ICS controlling MFW to the minimum steam generator level setpoint including uncertainty.

[ ]

Emergency Feedwater System

The three emergency feedwater (EFW) pumps automatically start upon a loss of both main feedwater pumps, and the two motor-driven pumps also start on a low steam generator level. Low main feedwater pump discharge pressure (ATWS Mitigation System Actuation Circuit) can also result in actuation of all three EFW pumps. If the ICS is functioning to throttle MFW flow by controlling on steam generator level, EFW is not modeled. For the cases that assume the ICS does not throttle MFW flow, EFW is actuated when the low MFW pump discharge pressure setpoint plus uncertainty is satisfied. Maximum EFW flow is assumed to maximize the cooldown. Nominally, the EFW flow is controlled to maintain a minimum steam generator level. The analysis assumes the EFW level control setpoint is higher and includes uncertainty. For the cases assuming a single failure in the EFW System, the EFW control valve to the affected steam generator is assumed to fail full-open. A conservatively low temperature is assumed for the EFW.

*Feedwater Isolation The steam line break analyses do not credit automatic isolation of main feedwater or emergency feedwater by the Automatic Feedwater Isolation System (AFIS).*

Turbine Control

The steam line break causes a rapid decrease in steam generator pressure. Thus, the ICS will attempt to close the turbine control valves in order to restore turbine header pressure to its setpoint. Since steam flow to the turbine is maximized if the turbine control valves remain open, it is conservative to assume that turbine control is in manual.

Turbine Bypass System

The Turbine Bypass System is assumed operable to limit the post-trip pressure in the unaffected steam generator, thereby minimizing the secondary-to-primary heat transfer from the unaffected steam generator to the RCS. This is conservative for maximizing the RCS cooldown.

15.3.1.1.5 With Offsite Power Results

The *limiting* steam line break with offsite power analysis assumes *uncontrolled main feedwater flow*. The single failure is assumed to be a train of Engineered Safety Features that results in only one train of HPI for the first 10 minutes. Table 15-1 gives the sequence of events for this case.

The steam line break initially causes the pressure to decrease in both steam generators (Figure 15-3). Break flowrates (Figure 15-4) for both steam generators rapidly increase. *Break flow from the affected steam generator then steadily decreases, following pressure, until liquid break flow occurs at 260 seconds due to overfilling of the steam generator.* After the turbine stop valves close, break flow from the unaffected steam generator stops, *and the unaffected steam generator repressurizes until about 30 seconds.* Beyond this point *in time, overcooling in* the affected steam generator decreases *pressure in the unaffected steam generator due to reverse heat transfer. Then a water-solid condition is reached in the unaffected steam generator* and it repressurizes. *The affected* steam generator *is* nearly fully depressurized by the end of the simulation. *The uncontrolled main feedwater flow overfills the affected steam generator at approximately 240 seconds, and the unaffected steam generator at 214 seconds.*

The cooldown in the affected loop *leads* the *cooldown in the* unaffected loop, as shown in the cold leg and hot leg temperature responses (Figure 15-5). The cold leg temperature in the unaffected loop increases once the turbine stop valves close, *and then the overcooling in the affected loop cools the unaffected loop.* The RCS has cooled to *less than 250*°F by the end of the simulation.

The total, moderator, Doppler, boron, and control rod reactivities are presented in Figure 15-6. The negative reactivity insertion at the beginning of the transient is due to the reactor trip and control rod insertion. The cooldown causes positive reactivity insertion due to the negative moderator and Doppler coefficients. The core *remains subcritical throughout the post-trip period, with the minimum subcritical margin reached at about 110 seconds. Boron injection from the core flood tanks, and then later from the HPI System provides sufficient negative reactivity to maintain the subcritical margin.*

The reactor power (Figure 15-7) decreases rapidly on reactor trip *and then approaches the decay heat power level.* The *minor* fluctuations in the thermal power heat flux are caused by flow surges in the core which result from flow degradation due to two-phase conditions in the unaffected loop. RCS pressure (Figure 15-8) rapidly decreases until the affected loop and reactor vessel head begin to saturate at approximately 4 seconds. After this time, RCS pressure continues to decrease for the remainder of the simulation, allowing the core flood tanks to inject.

Core inlet mass flow (Figure 15-9) initially increases with time following the steam line break. Since the reactor coolant pumps provide essentially constant volumetric flow, the decreasing RCS temperatures initially result in an increase in mass flow. However, as the unaffected loop begins to void and RCP performance degrades as predicted by the RETRAN two-phase pump degradation model, core inlet flow decreases to approximately *80%* of the initial flow. *Core flood tank and HPI System injection refill the RCS, and single-phase flow is restored. by 160 seconds.*

15.3.1.2 VIPRE-01 Analysis

15.3.1.2.1 Initial and Boundary Conditions

The RETRAN-02 analyses provide the limiting statepoint core exit pressure, core inlet temperature, core inlet flow rate, and core average heat flux for [ ]. These boundary conditions are input to VIPRE-01 as steady state boundary conditions.

15.3.1.2.2 Axial and Radial Power Distributions

<u>Axial Power Distributions</u>

[ ]

## Radial Power Distribution

The maximum pin radial power peak in the hot assembly is calculated explicitly by SIMULATE. Also, utilizing the hot assembly pin radial power distributions as described in Reference 15-2, the hot assembly pin radial power distributions for the return-to-power situation can be derived. For [ ]

### 15.3.1.2.3 Flow Correlations

For the steam line break with offsite power case, subcooled and bulk voids are modeled with the [ ] Sensitivity studies have shown that using this combination of void correlations results in an acceptable prediction of DNBR.

### 15.3.1.2.4 Conservative Factors

Conservative factors described in Reference 15.2 are applied to the [ ] channel VIPRE-01 model. These conservative factors are the hot channel area reduction factors (2% for the hot unit subchannel and 3% for the hot instrumentation subchannel), the engineering hot channel factor ($F_q$) of 1.013, and the core inlet flow maldistribution factor. Based on the vessel model flow test and Oconee core pressure drop measurement, the core inlet flow maldistribution is conservatively modeled as a reduction in the hot assembly flow. Since in the with offsite power RETRAN-02 analysis two RCPs are assumed to trip, the hot assembly flow reduction factor for the VIPRE-01 DNB analysis is therefore [ ]% as described in Section 9.3.2.3.

15.3.1.2.5 Critical Heat Flux Correlation

The W-3S CHF correlation is used for the with offsite power steam line break DNBR analysis. The historical range of applicability for the W-3S correlation is (Reference 15-3):

| | |
|---|---|
| Pressure (psia) | 1000 to 2300 |
| Mass flux ($10^6$ lbm/hr-ft$^2$) | 1.0 to 5.0 |
| Quality (equilibrium) | -0.15 to 0.15 |

The W-3S CHF correlation has been approved by the NRC for analysis with system pressures as low as 500 psia and mass flux as low as 0.5 x $10^6$ lbm/hr-ft$^2$ (References 15-9 and 15-10).

*The BWU-Z CHF correlation with the 0.98 Mk-B11 multiplier is also used for the with offsite power steam line break DNBR analysis for Mk-B11 fuel only. The minimum DNBR design limits are as follows for the following parameter ranges of applicability for this correlation and design limit:*

| | | *Design DNBR* |
|---|---|---|
| *Pressure (psia)* | *315 to 700* | *1.59* |
| | *700 to 1000* | *1.20* |
| | *1000 to 2465* | *1.19* |
| *Mass flux ($10^6$ lbm/hr-ft$^2$)* | *0.36 to 3.55* | |
| *Quality* | *less than 0.74* | |

15.3.1.2.6 Results

*Using the limiting statepoint from the RETRAN-3D analysis discussed in Section 15.3.1.1.5 it was evident that the reactor remained subcritical following reactor trip. Therefore, no detailed VIPRE analysis was necessary and the acceptance criterion discussed in Section 15.1.2, is met.*

15.3.1.3 SIMULATE-3P Analysis

The limiting RETRAN statepoint conditions for the steam line break analysis with offsite power are input to SIMULATE-3P. The SIMULATE analysis also demonstrates that a *large negative post-trip* reactivity margin is maintained. Therefore, the RETRAN reactivity prediction is conservative. *The results show that the reactor remains well critical after reactor trip, and so the SIMULATE-3P power distribution results do not need to be input to the VIPRE-01 code.*

### 15.3.2 Without Offsite Power

#### 15.3.2.1 RETRAN-02 Analysis

##### 15.3.2.1.1 Initial Conditions

The initial conditions for the steam line break analysis without offsite power are selected to maximize the RCS depressurization and maximize the post-trip core power response. The steam line break analysis without offsite power is very similar to a loss of coolant flow analysis (Chapter 9.0). Thus, sensitivity study results from the loss of coolant flow analysis are utilized to select appropriate initial conditions. The transient RCS conditions for the steam line break without offsite power are within the ranges covered by the statistical core design (SCD) approach. Therefore, the analysis will utilize the SCD approach.

Power Level
Nominal full power will be assumed since the uncertainty in power is accounted for in the SCD limit.

RCS Pressure
Low initial pressure is generally conservative for DNB calculations. The SCD limit accounts for the uncertainty in indicated pressure.

Pressurizer Level
Sensitivity studies have concluded that initial pressurizer level is not an important parameter with respect to DNB for the steam line break with offsite power lost analysis.

RCS Temperature
Nominal RCS average temperature will be assumed. The indication uncertainty and ICS deadband associated with T-ave are accounted for in the SCD limit.

RCS Flow

A low initial flow rate is conservative with respect to DNB calculations. The uncertainty in RCS flow is accounted for in the SCD limit.

Core Bypass Flow

A high core bypass flow is assumed to minimize the coolant flow along the fuel rods.

Fuel Temperature

A high initial fuel temperature is conservative with respect to DNB calculations for loss of flow analyses. Since BOC kinetics parameters are assumed, a maximum BOC fuel temperature is assumed.

Steam Generator Mass

A conservatively high steam generator mass is assumed to maximize the overcooling.

#### 15.3.2.1.2 Boundary Conditions

For a steam line break with coincident loss of offsite power, the reactor will trip and the RCPs will begin to coast down. For this scenario the accident resembles a loss of flow accident with a coincident depressurization. For a loss of flow accident, the minimum DNBR statepoint is expected within the first few seconds of the RCP coastdown. Therefore, detailed modeling of many boundary conditions that would not occur until after the limiting statepoint are unnecessary. The boundary conditions for the steam line break with offsite power lost which differ from the with offsite power case are as follows:

Loss of Offsite Power

The loss of offsite power occurs coincident with the break. The control rods are assumed to lose power coincident with the loss of offsite power. Upon losing power, control rod insertion is delayed to account for gripper coil release delay. The loss of offsite power also initiates a coastdown of the RCPs.

Decay Heat

A decay heat multiplier curve is applied to the 1979 ANS Standard 5.1 decay heat power to ensure that the RETRAN prediction of decay heat is conservatively maximized. Maximum decay heat is conservative for loss of flow DNB analyses.

Single Failure

No single failure could be identified which affects the results.

15.3.2.1.3 Physics Parameters

Moderator Temperature Coefficient

Reactivity insertion curves as a function of temperature are used to model moderator temperature feedback. BOC least negative values are conservative. This assumption minimizes the negative feedback associated with any core moderator heatup that occurs with a loss of flow.

Doppler Temperature Coefficient

Reactivity insertion curves as a function of temperature are used to model Doppler temperature feedback. BOC least negative values are conservative. This assumption minimizes the negative feedback associated with any fuel heatup that occurs with a loss of flow.

Beta-effective and Neutron Lifetime

A large $\beta_{eff}$ and prompt neutron lifetime are chosen to slow the core power decrease on control rod insertion. BOC decay constants and delayed neutron precursor fractions are also utilized.

Scram Curve and Worth

The control rods are inserted when offsite power is lost. For this analysis, a bottom-peaked scram curve and an upper bound on the rod insertion time are assumed. These assumptions maximize the post-trip energy addition, which is conservative for the DNB prediction. A minimum trippable worth (not to exceed a 1% $\Delta$/k subcritical margin), including an allowance for the most reactive rod stuck out of the core, is utilized in the analysis.

15.3.2.1.4 Control, Protection, and Safeguards Systems

Main Feedwater System

On a loss of offsite power, the hotwell pumps and condensate booster pumps will trip, resulting in a trip of the main feedwater pumps on low suction pressure. With the suction head diminishing, the MFW pumps will rapidly coastdown. A maximum coastdown time is assumed for the MFW pumps.

Emergency Feedwater System

The Emergency Feedwater System cannot start and deliver flow in the short duration of this analysis and is not modeled.

*Feedwater Isolation* *The steam line break analyses do not credit automatic isolation of main feedwater or emergency feedwater by the Automatic Feedwater Isolation System (AFIS).*

15.3.2.1.5 Results

The steam line break without offsite power case assumes offsite power is lost coincident with the opening of the steam line break. Thus, an RCS flow coastdown also begins with the opening of the break. Table 15-2 gives the sequence of events for this case.

The steam line break initially causes the pressure to decrease in both steam generators (Figure 15-11). Once the turbine stop valves close, the unaffected steam generator repressurizes. The affected steam generator has depressurized to about *750* psig by the end of the simulation. The break flow response is similar to what has been discussed for the with offsite power analysis. The cooldown in the affected loop is almost the same as in the unaffected loop during the first five seconds, as shown in the cold leg temperature response (Figure 15-12). The affected loop hot leg temperature is slightly higher than the unaffected loop hot leg temperature due to the outsurge of hot liquid from the pressurizer. The slight increase in hot leg temperatures from 2 to 5 seconds can be attributed to the RCS flow coastdown.

The RCS volumetric flow decreases for the duration of the simulation (Figure 15-13). This is the result of the loss of offsite power. The loss of offsite power also results in control rod insertion, which drives the core kinetics response (Figure 15-14). Due to control rod insertion, the core average fuel temperature begins to decrease. However, due to the relatively slow changes in the moderator and fuel temperatures, and given that the time period of interest for DNB is within the first 1-2 seconds of the flow coastdown, the moderator and Doppler feedback for the offsite power lost analysis are generally negligible.

The reactor power decreases rapidly on reactor trip (Figure 15-15). The core thermal power also decreases after reactor trip, but does not decrease as fast as neutron power. RCS pressure (Figure 15-16) initially decreases due to the effects of the steam line break and control rod insertion. As flow and primary-to-secondary heat transfer begin to degrade, the RCS pressure *begins to recover between 3 to 5 seconds*. The RCS pressure increase is also a result of the closure of the turbine stop valves.

### 15.3.2.2 VIPRE-01 Analysis

#### 15.3.2.2.1 Initial and Boundary Conditions

The RETRAN analyses provide the transient core exit pressure, core inlet temperature, core inlet flow rate, and core average heat flux for both core halves of the split reactor vessel model. For the without offsite power analysis, both core halves have identical transient boundary conditions for the duration of the analysis. These boundary conditions are input to VIPRE as transient forcing functions.

#### 15.3.2.2.2 Axial and Radial Power Distributions

For the SCD statepoint analysis, the axial power distribution is a chopped cosine shape with an axial peak of [ ] peaked at X/L = [ ], and the radial power distribution is the base model radial power distribution with a hot pin radial power of [ ] (Reference 15-3). For the maximum allowable radial peak (MARP) analyses, a set of axial power shapes are analyzed. The magnitude and elevation of the axial shape is varied to cover the full range of shapes resulting from the nuclear design analysis.

15.3.2.2.3 Conservative Factors

Since the SCD methodology is utilized for predicting the DNBR, the SCD limit accounts for most of the uncertainties in key parameters. Based on the vessel model flow tests and Oconee core pressure drop measurement, the core inlet flow maldistribution is conservatively modeled as a reduction in the hot assembly flow. The hot assembly flow reduction factor for four-pump operation is 5%.

15.3.2.2.4 Critical Heat Flux Correlation

*The BWU-Z critical heat flux (CHF) correlation is used for the steam line break transient DNBR analysis for the results presented. The range of applicability for the BWU-Z CHF correlation using the SCD procedure is:*

| | |
|---|---|
| *Pressure (psia)* | *1600 to 2242* |
| *Mass flux (Mlbm/hr-sqft)* | *0.36 to 3.55* |
| *Quality* | *< 0.74* |

*The BWU-Z CHF correlation SCD limit for the steam line break transient is determined utilizing the minimum DNBR statepoint boundary conditions described in Section 15.3.2.2.5.*

15.3.2.2.5 Results

The transient VIPRE DNBR results are shown in Figure 15-17, with a minimum DNBR of *1.90 at 1.51* seconds. This statepoint is used to determine the SCD limit for the steam line break transient. The MARP results are shown in Figure 15-18.

15.3.2.3 <u>Fuel Pin Census</u>

The MARPs are used for the fuel pin census. When the radial power peak of the fuel pin exceeds the MARP limit during the transient, DNB and cladding failure are assumed to occur. The fuel pin census is performed to determine the number of failed fuel pins during the steam line break accident. The results of the fuel pin census indicate that no peaks exceed the MARP limits, and therefore no cladding failure occurs for the steam line break accident. Based on this result the core will remain intact for effective core cooling.

## 15.5 References


15-1 RETRAN-02: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 4, EPRI, November 1988

15-2 *Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002*

15-3 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Revision 3, EPRI, August 1989

15-4 *Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Duke Power Company, February 24, 1993*

15-5 *Mass and Energy Release and Containment Response Methodology, DPC-NE-3003-P, Revision 1, Duke Power Company, June 2002*

15-6 BWC Correlation of Critical Heat Flux, BAW-10143P-A, April 1985

15-7 BAW-10199-PA, The BWU Critical Heat Flux Correlations, Addendum 1, September 1996

15-8 Letter, D. E. LaBarge (NRC) to W. R. McCollum (Duke), SER on topical report DPC-NE-3000-PA, Revision 2, October 14, 1998

15-9 Letter, A. S. Thadani (NRC) to W. J. Johnson (Westinghouse), SER on WCAP-9226-P, Reactor Core Response to Excessive Secondary Steam Releases", January 31, 1989

15-10 Letter, T. A. Reed (NRC) to H. B. Tucker (Duke), SER on topical report DPC-NE-3001, November 15, 1991

Table 15-1

Sequence of Events

Steam Line Break - With Offsite Power

| Event | Time (sec) |
|---|---|
| Break opens | 0.0 |
| Third CBP starts | *0.5* |
| Reactor trip on variable low pressure-temperature | *3.1* |
| Control rod insertion begins | *3.2* |
| Turbine stop valves closed<br>Control rods fully inserted | *4.2* |
| HPI actuates | *35.5* |
| CFT injection begins | *97.4* |
| Boron from CFT B starts / *time of minimum subcritical margin* | *110.1* |
| Boron from CFT A starts | *115.1* |
| Boron injection from HPI begins | *115.8* |
| Simulation ends | 600.0 |

Table 15-2

Sequence of Events

Steam Line Break - Without Offsite Power

| Event | Time (sec) |
|---|---|
| *Break Opens, LOOP Occurs* | *0.0* |
| *RCPs Begin Coastdown* | |
| *Reactor Trips* | |
| *Control Rod Insertion Begins* | *0.14* |
| *Turbine Stop Valves Closed* | *1.72* |
| *Control Rods Fully Inserted* | *2.54* |
| *Simulation Ends* | *5.0* |

Figure 15-3
## LARGE STEAM LINE BREAK
WITH OFFSITE POWER

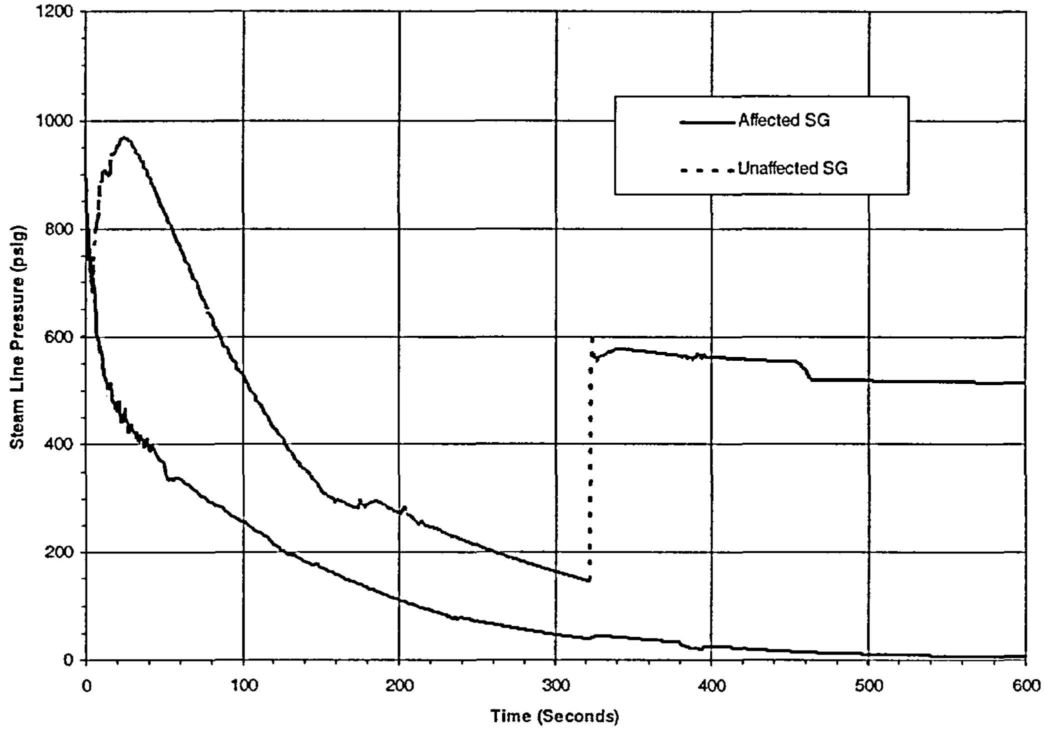


Figure 15-4
## LARGE STEAM LINE BREAK
WITH OFFSITE POWER

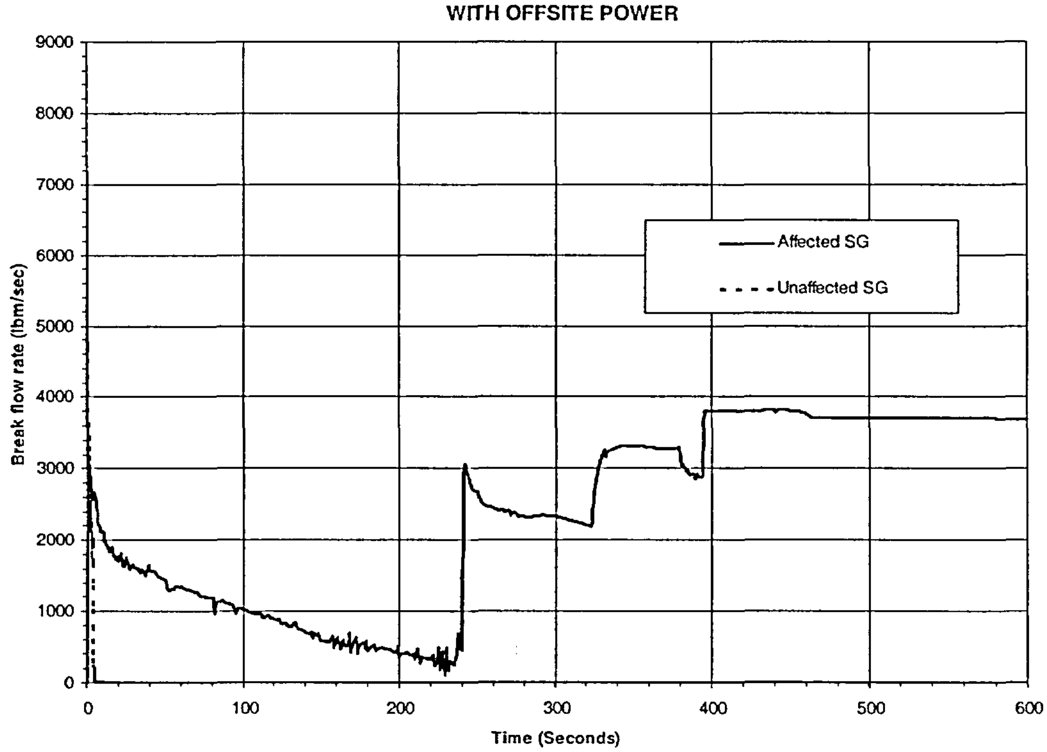

Figure 15-5
**LARGE STEAM LINE BREAK**
**WITH OFFSITE POWER**

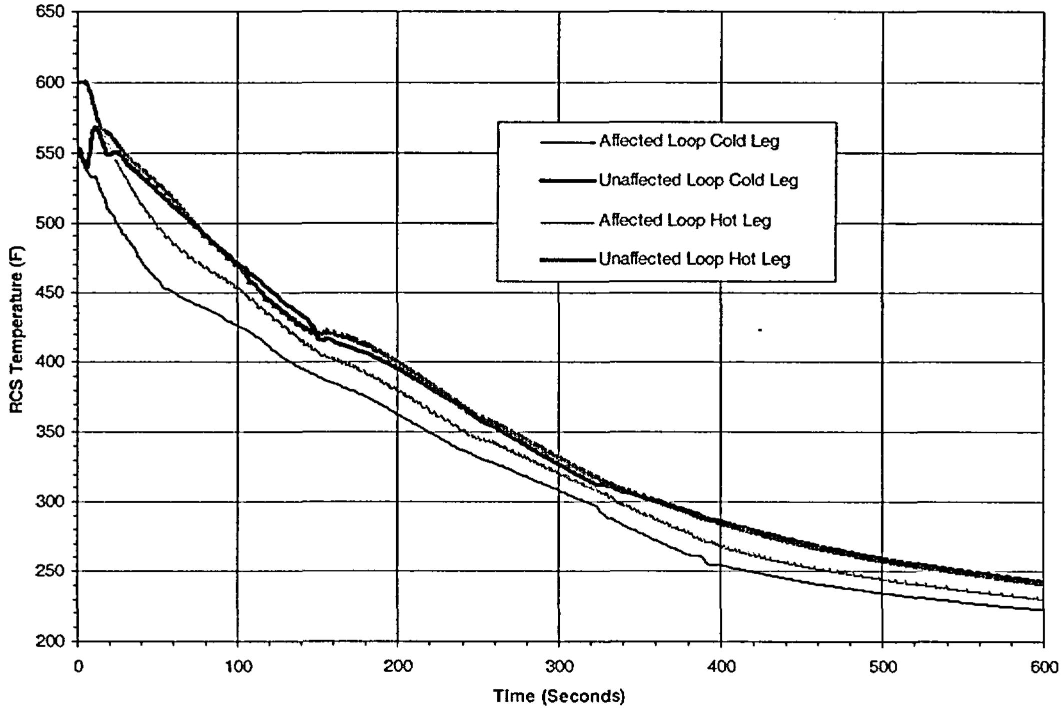


Figure 15-6
**LARGE STEAM LINE BREAK**
**WITH OFFSITE POWER**

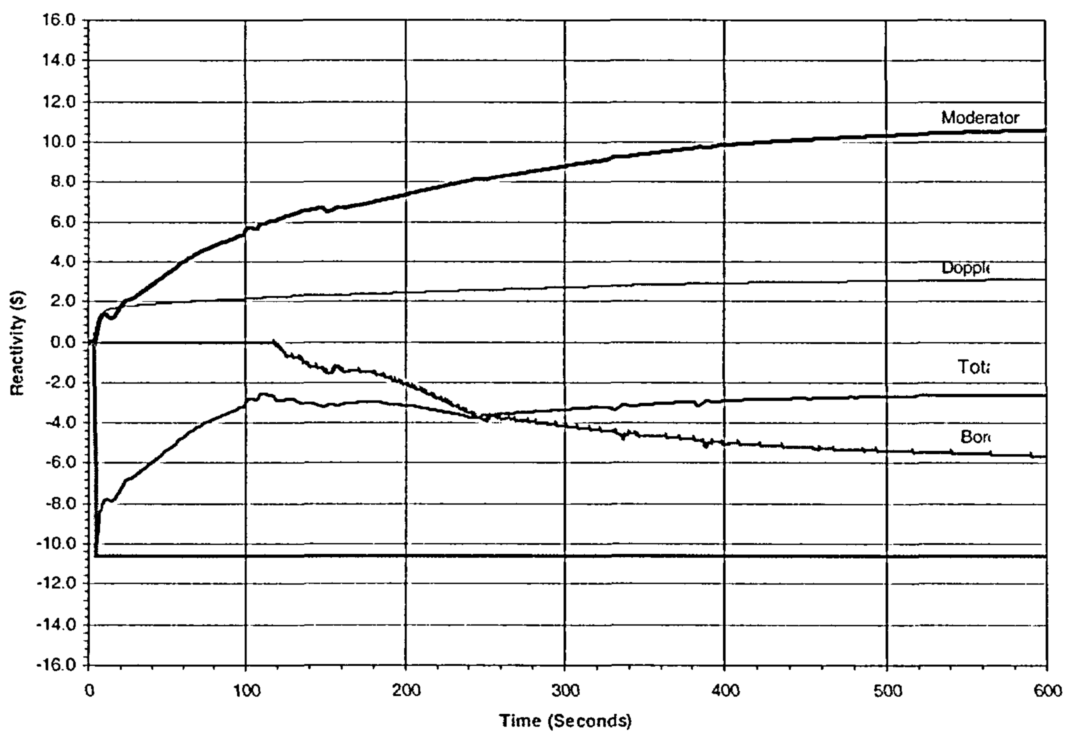

Figure 15-7

## LARGE STEAM LINE BREAK

WITH OFFSITE POWER

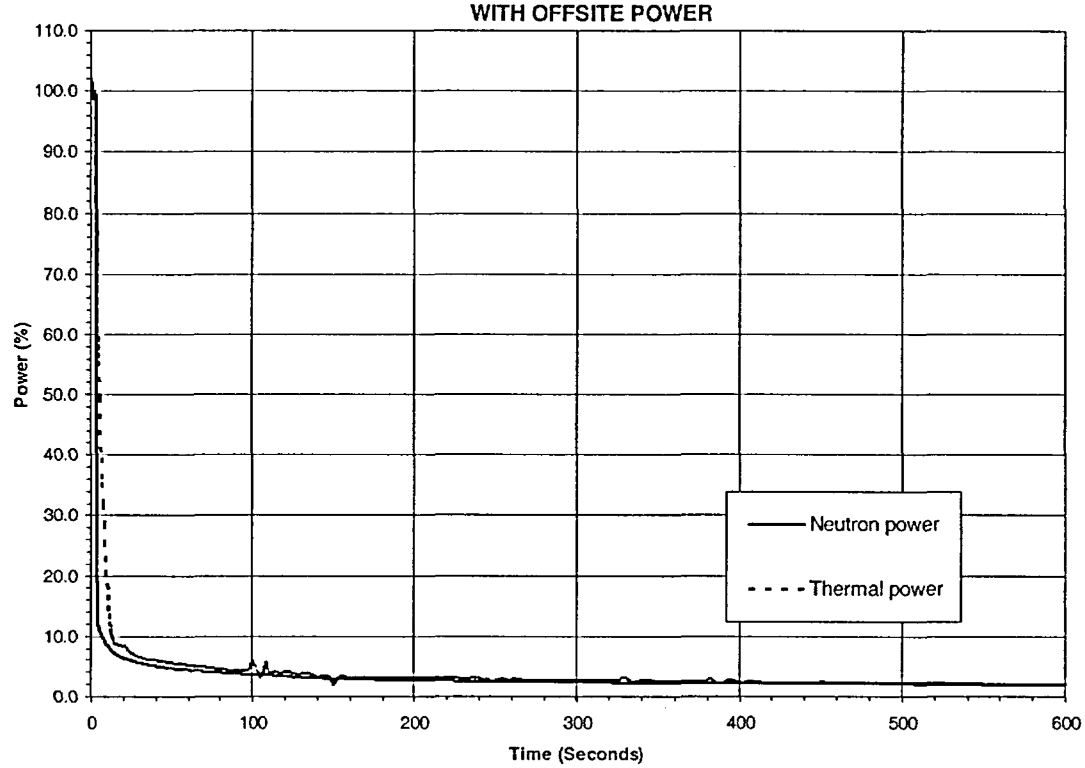


Figure 15-8

## LARGE STEAM LINE BREAK

WITH OFFSITE POWER

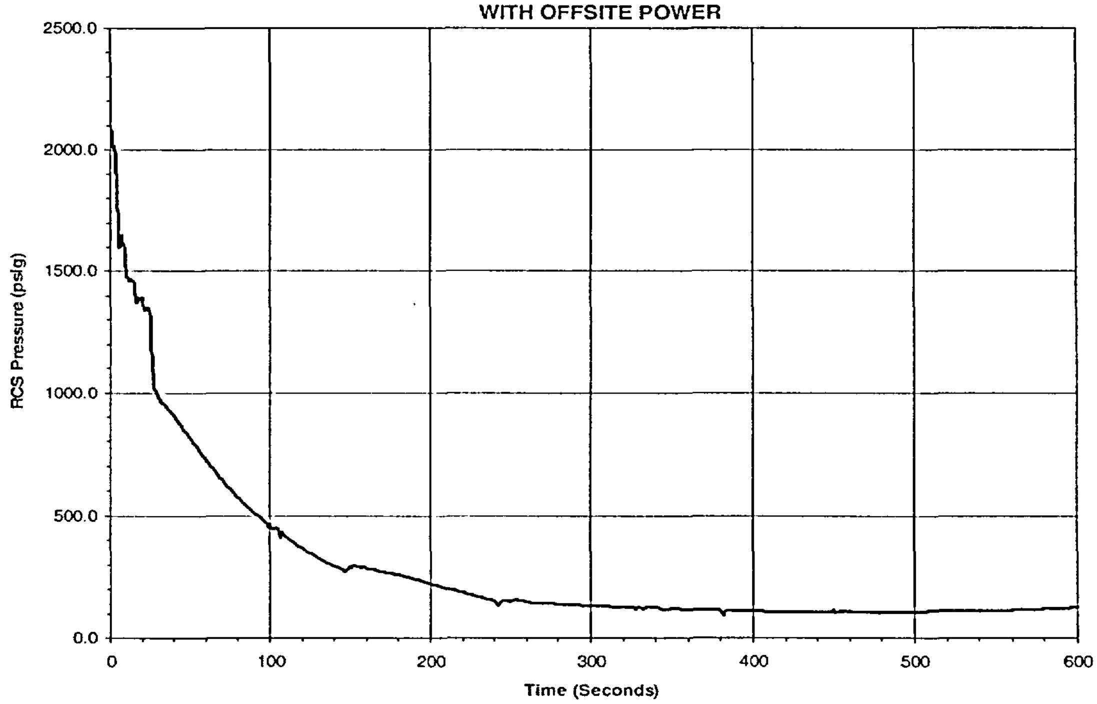

Figure 15-9

LARGE STEAM LINE BREAK

WITH OFFSITE POWER

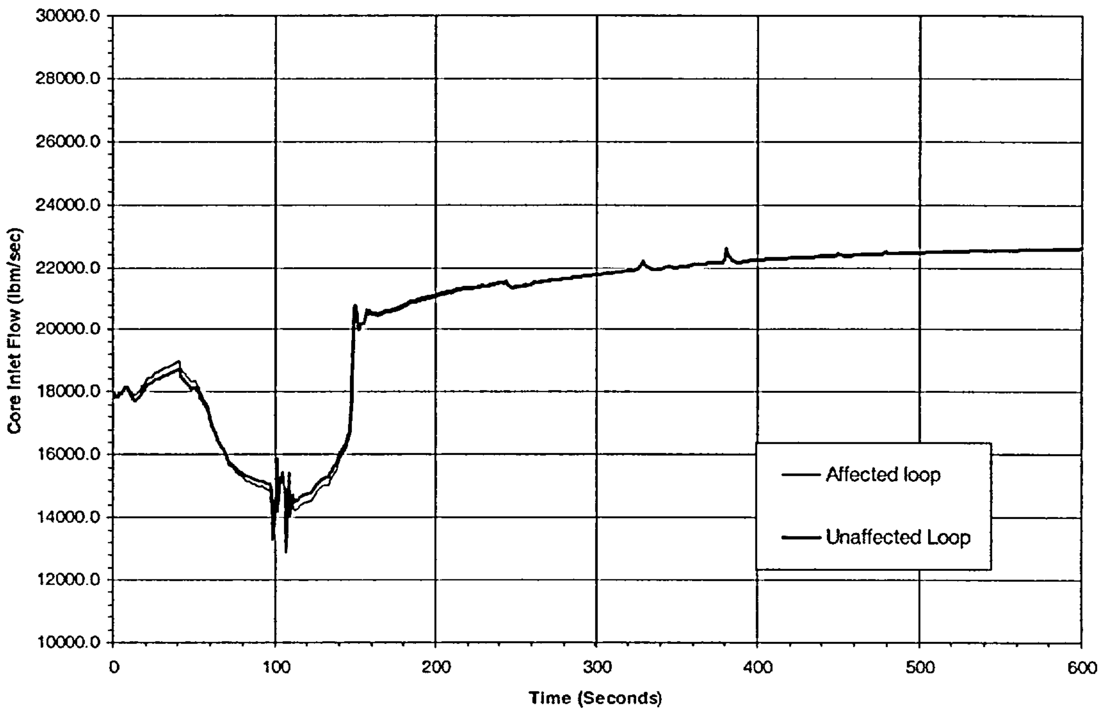

Figure 15-11
## LARGE STEAM LINE BREAK
WITHOUT OFFSITE POWER

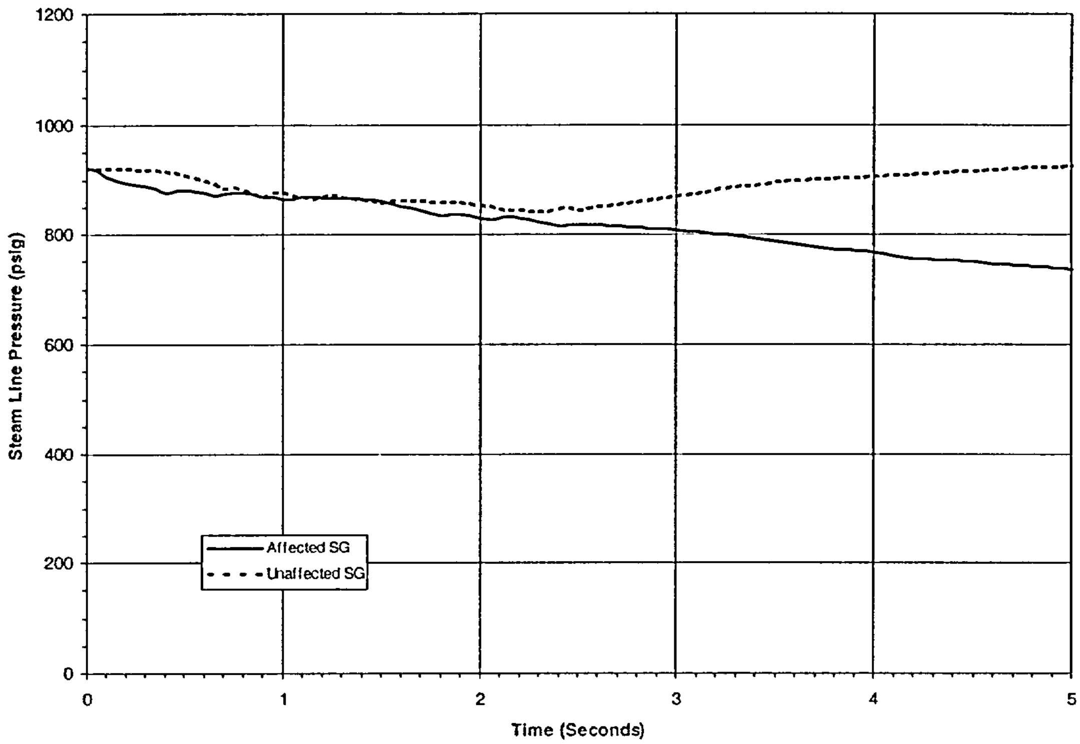


Figure 15-12
## LARGE STEAM LINE BREAK
WITHOUT OFFSITE POWER

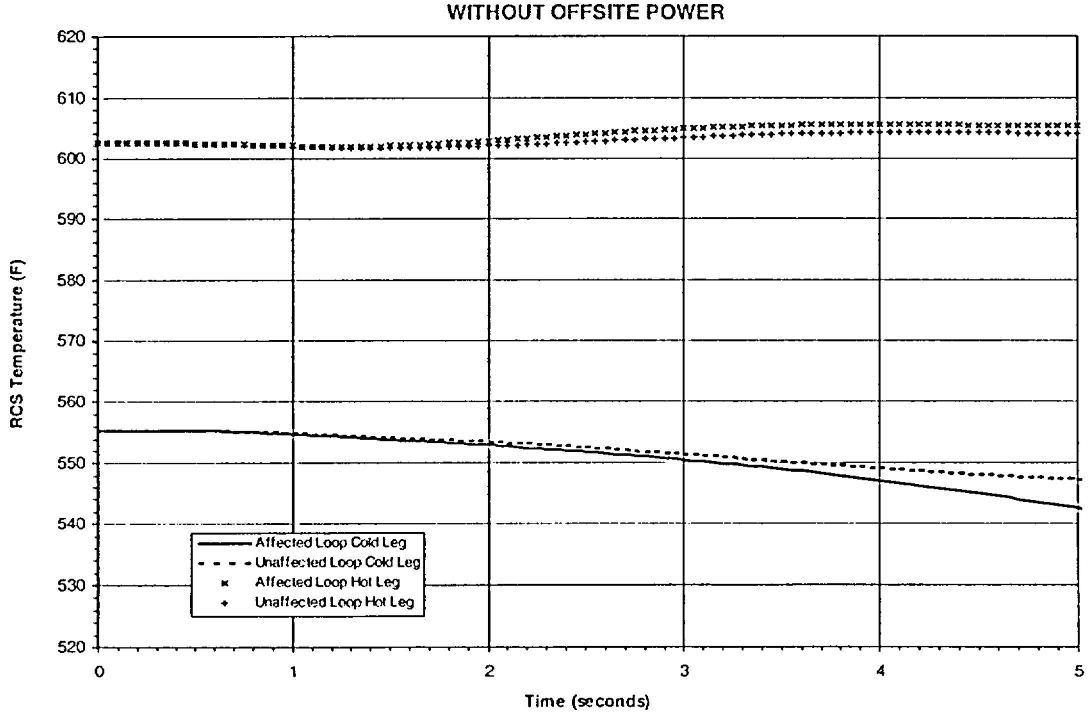

Figure 15-13
**LARGE STEAM LINE BREAK**
WITHOUT OFFSITE POWER

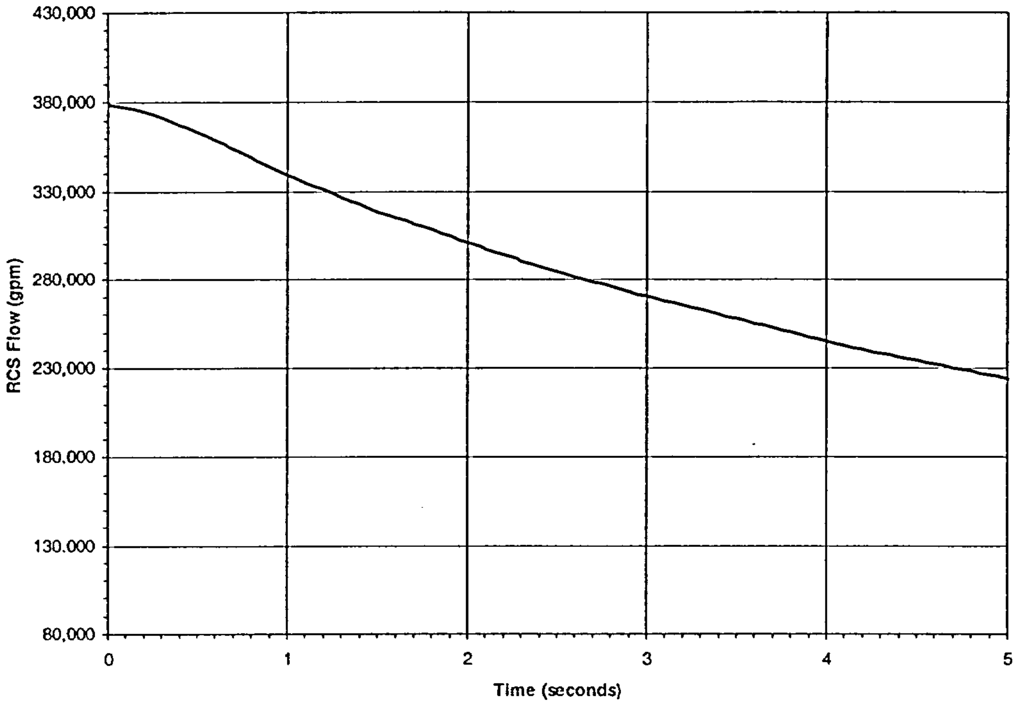


Figure 15-14
**LARGE STEAM LINE BREAK**
WITHOUT OFFSITE POWER

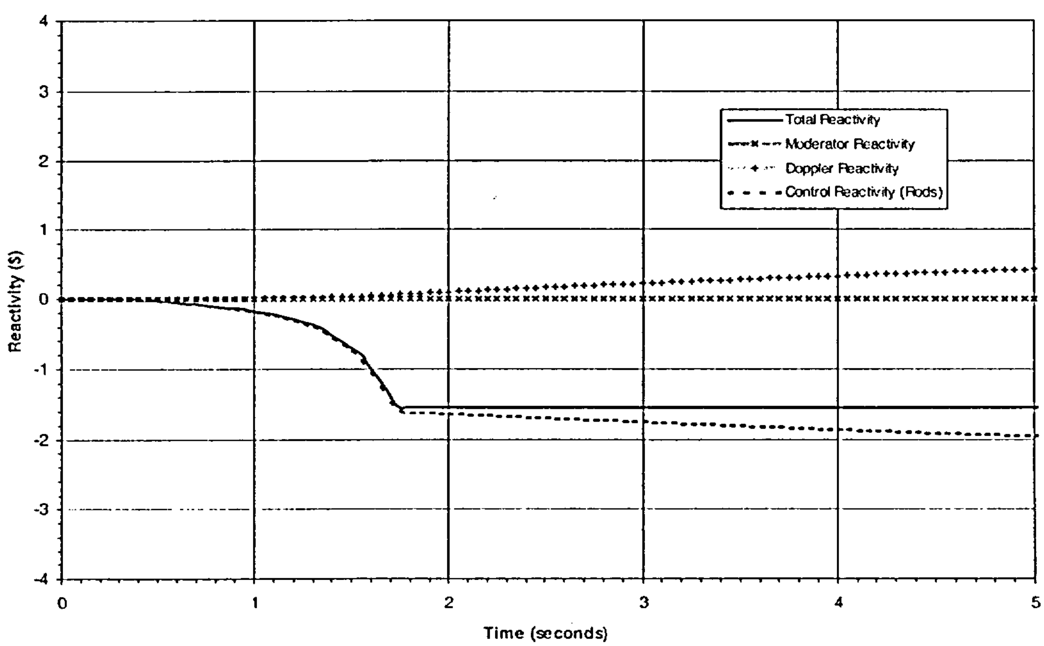

Figure 15-15
LARGE STEAM LINE BREAK
WITHOUT OFFSITE POWER

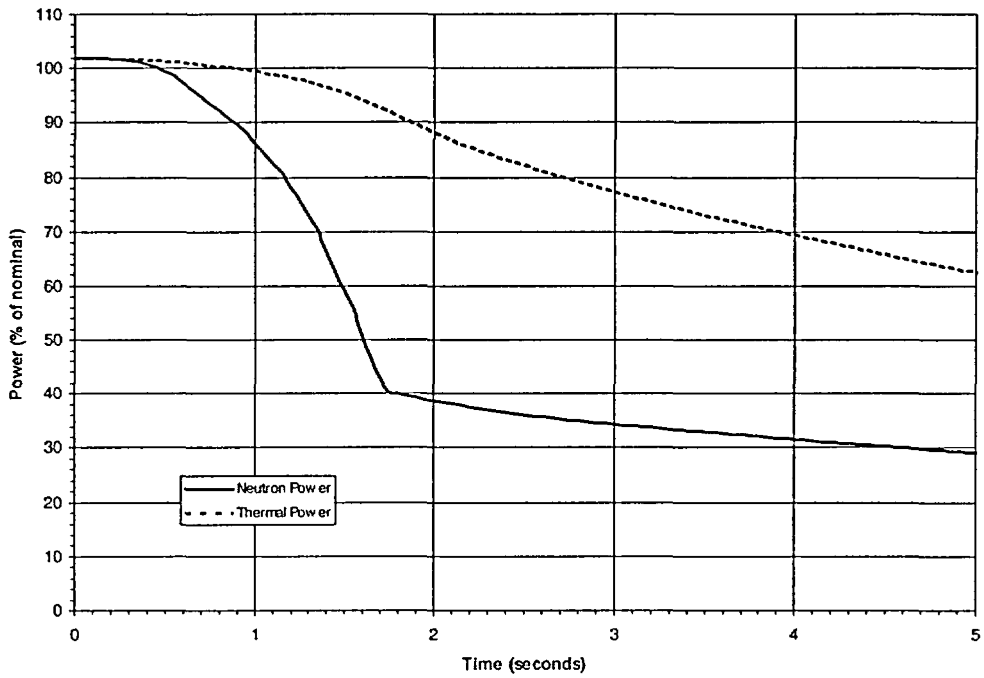


Figure 15-16
LARGE STEAM LINE BREAK
WITHOUT OFFSITE POWER

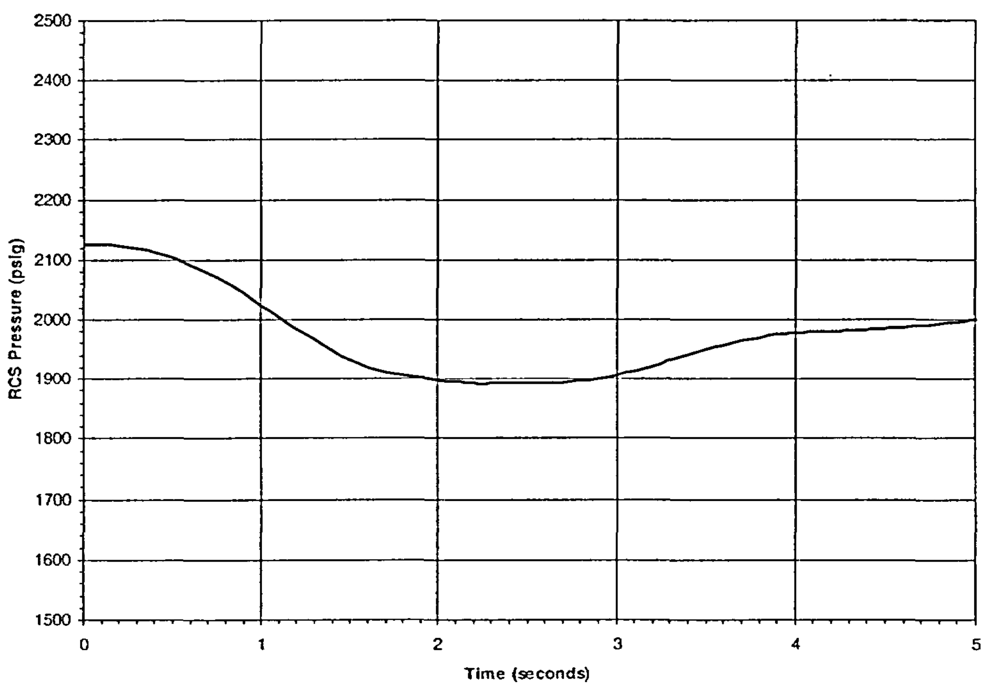

Figure 15-17
LARGE STEAM LINE BREAK
WITHOUT OFFSITE POWER

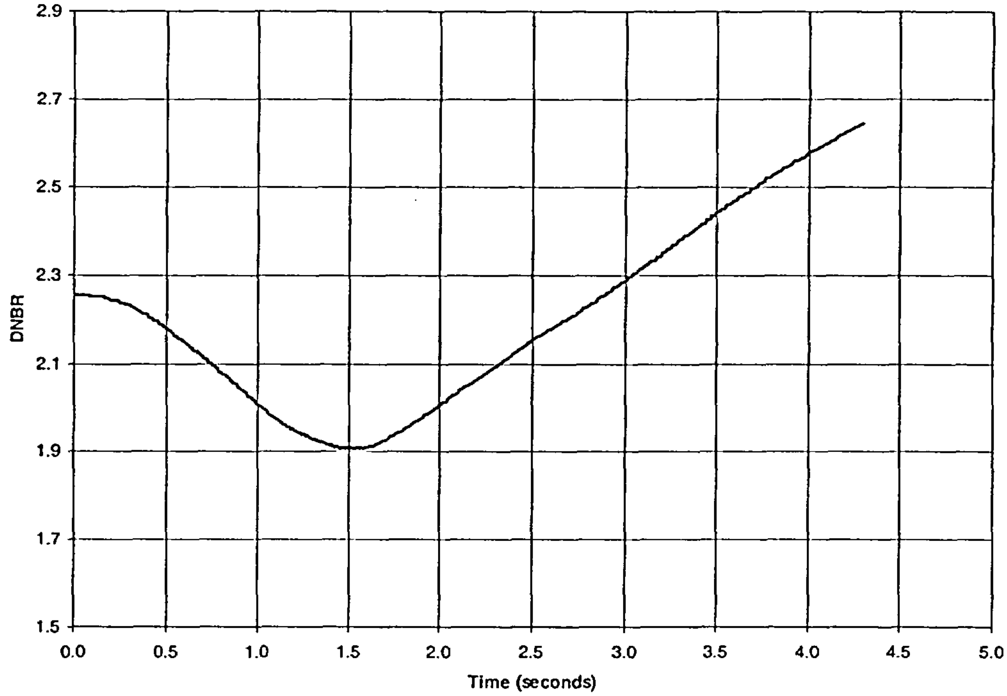


Figure 15-18
LARGE STEAM LINE BREAK
WITHOUT OFFSITE POWER

[ ]

# Attachment 4

## Additional Revisions To Topical Report DPC-NE-3000, Revision 3 "Thermal-Hydraulic Transient Analysis Methodology"

The following additional revisions to topical report DPC-NE-3000-P, Revision 3, "Thermal-Hydraulic Transient Analysis Methodology," are submitted for NRC review and approval. These revisions to the topical report were identified subsequent to the submittal of Revision 3 on June 13, 2002.

<u>Chapter 2 - Oconee Transient Analysis Revisions</u>

1. Section 2.1.5.4, MFW and EFW Automatic Isolation, revise to only describe the AFIS design since it has now been installed on all three units. Refer to Attachment 1, Item 9 of the June 13, 2002 submittal.

   Replace Section 2.1.5.4 with the following:

   *The Automatic Feedwater Isolation System (AFIS) uses low steam generator outlet pressure and turbine header pressure as the initiating parameter for automatic MFW isolation. Both MFW pumps are tripped, but the MFW valves close only to a steam generator with a low pressure condition. The turbine-driven EFW pump is stopped. The motor-driven EFW pump to each steam generator is stopped if the depressurization rate in that steam generator exceeds the setpoint, and the low pressure MFW isolation has occurred. Thus, the AFIS design includes automatic isolation of EFW flow to a depressurizing steam generator.*

   Technical Justification: The AFIS design is now installed in all three Oconee units. This is an editorial revision to maintain consistency with the plant design.

2. Section 2.2.6.2, Centrifugal Pumps, is revised to change the reactor coolant pump two-phase degradation modeling.

   Change: ". . . *by applying two-phase multipliers from Reference 2-3, Tables VI.1-4 and VI.1-5, to the built in Bingham homologous curves.*"

   To: ". . . *through the use of two-phase multipliers and two-phase difference curves developed from CE-EPRI test data (Reference 2-15).*

   *2-15 W. G. Kennedy, et. Al., "Pump Two-Phase Performance Program: Volumes 1-8", EPRI NP-1556, September 1980*"

   Technical Justification: The RETRAN reactor coolant pump two-phase head and torque degradation model input has been revised to use the CE-EPRI test data per the above reference. This data is considered more applicable to the Oconee reactor coolant pumps than the previous data that was based on the Semiscale facility pump data.

Chapter 3 - McGuire/Catawba Transient Analysis Revisions

3. Section 3.2.6.2, Centrifugal Pumps, is revised to change the reactor coolant pump two-phase degradation modeling.

Change: "... *by applying two-phase multipliers from Reference 3-3, Tables VI.1-4 and VI.1-5, to the homologous curves described above.*"

To: "... *through the use of two-phase multipliers and two-phase difference curves developed from CE-EPRI test data (Reference 3-15).*

*3-15 W. G. Kennedy, et. Al., "Pump Two-Phase Performance Program: Volumes 1-8", EPRI NP-1556, September 1980*"

Technical Justification: The RETRAN reactor coolant pump two-phase head and torque degradation model input has been revised to use the CE-EPRI test data per the above reference. This data is considered more applicable to the McGuire and Catawba reactor coolant pumps than the previous data that was based on the Semiscale facility pump data.

Appendix A - Methodology Revisions for Mk-B11 Fuel Revisions

4. Appendix A, page A-2, Critical Heat Flux Correlation, the DNBR design limits for the BWU-Z CHF correlation need to be corrected.

Replace the second paragraph with the following:

*"The design DNBR limit using the VIPRE-01 code has been determined as a function of pressure ranges for Mk-B11 fuel as stated in Reference A-4.*

| *Pressure (psia)* | *Design DNBR* |
|---|---|
| *400 to 700* | *1.59* |
| *700 to 1000* | *1.20* |
| *1000 to 2465* | *1.19* |

*The statistical core design methodology (SCD) of Reference 2-14 may also be used. The SCD methodology is generally used unless the analysis parameters are not bounded by the ranges considered in the methodology. A typical SCD limit using VIPRE-01 with the BWU-Z correlation and the Mk-B11 multiplier of 0.98 is 1.33 for pressures above [ ] psi."*

Technical Justification - The current text does not accurately explain that the design DNBR limit varies with the BWU-Z correlation as a function of the pressure range. The above changes correct this oversight. Also, the 0.98 value of the Mk-B11 multiplier is given. These are editorial changes only.

## Attachment 6

## Additional Revisions To
## Topical Report DPC-NE-3005, Revision 2
## "UFSAR Chapter 15 Transient Analysis Methodology"

The following additional revisions to topical report DPC-NE-3005, Revision 2, "UFSAR Chapter 15 Transient Analysis Methodology," are submitted for NRC review and approval. These revisions to the topical report were identified subsequent to the submittal of Revision 2 on June 13, 2002.

Chapter 1 - Introduction and Summary Revisions

1. Section 1.3, Analysis Methodology, Main Feedwater Isolation" (Refer to Attachment 5, Item #9 in June 13, 2002 submittal), the text is revised to reflect the station as-built status for the AFIS modification

   Change: "*The large and small steam line break analyses do not credit automatic isolation of main feedwater. Automatic isolation of main feedwater is by the Automatic Feedwater Isolation System (AFIS) for Unit 1, and by the Main Steam Line Break Detection and Feedwater Isolation Circuitry for Units 2 and 3 (until AFIS is installed on those units).*"

   To: "*The large and small steam line break analyses do not credit automatic isolation of main feedwater or emergency feedwater by the Automatic Feedwater Isolation System (AFIS).*

   Technical Justification: The AFIS System has now been installed on all three units. This editorial change is being made to keep the topical report consistent with current design.

Chapter 11 - Dropped Rod Analysis Methodology Revisions

2. Section 11.1.1, Nodalization, is revised to describe additional modeling details for the main steam lines and condenser.

   Delete the following sentence:

   *"A junction is added to the base model to connect the steam lines since an asymmetric steam generator response will occur during cases with three-pump operation."*

   Insert the following text:

[ ]

Technical Justification: The original text does not completely describe the base model nodalization changes that are used in the dropped rod analysis. The revised text includes all of the nodalization changes. The revised text is similar to the Chapter 16 text for the small steam line break analysis since the same modeling is used for both events.

Chapter 15 - Steam Line Break Methodology Revisions

3. Section 15.2.1.5, Replacement Steam Generator RETRAN-3D Model, add a sentence regarding increasing the junction flow area for the steam generator cold leg outlet nozzle in the unaffected loop is increased if a code abort occurs.

   Insert the following at the end of the paragraph:

   *"The junction flow area for the steam generator cold leg outlet nozzle in the unaffected loop is increased if a code abort occurs."*

   Technical Justificaton: During certain RETRAN-3D steam line break analyses a code abort occurs. The cause of this problem was determined by the code vendor to be a problem with the momentum flux term in the momentum equation at the steam generator cold leg nozzle when voiding occurs. Several options to work around this error were discussed. It was decided that the best option was to increase the junction flow area during a restart to decrease the momentum flux term. This approach was successful in avoiding the job abort. This modeling approach is only used when a job abort occurs. No adverse impacts on the simulation results were identified as a result of this modeling approach.

4. Section 15.2.1.6, Steam Generator Water Carryout Control, has been revised for the main steam line break modeling for the replacement steam generators using RETRAN-3D.

   Insert the following at the beginning of the second paragraph:

   *"The following modeling approach is used for the RETRAN-02 analysis of the original steam generators."*

   Insert the following new third paragraph:

Technical Justification:

5. Section 15.3.1.1.2, Reactor Coolant Pump Modeling, the text is revised to describe the current approach to tripping the reactor coolant pumps.

Change: "*The RCPs in the unaffected loop are tripped at 100 seconds to avoid a code error associated with pressure oscillations in this loop due to two-phase performance. Tripping the RCPs in the unaffected loop has a conservative impact on the cooldown of the RCS since there is reverse heat transfer taking place in the steam generator.*"

To: "*In the RETRAN-02 analysis for the original steam generators, the RCPs in the unaffected loop are tripped at 100 seconds to avoid a code error associated with pressure oscillations in this loop due to two-phase performance. Tripping the RCPs in the unaffected loop has a conservative impact on the cooldown of the RCS since there is reverse heat transfer taking place in the steam generator." In the RETRAN-3D analysis for the replacement steam generators this code error does not occur and the RCPs are not tripped."*

Technical Justification: In the RETRAN-02 analysis a code abort occurred and tripping the RCPs in the unaffected loop was the approach taken to avoid the code abort. This approach was judged to be a conservative work around. In the RETRAN-3D analysis this code abort does not occur, and so tripping the RCPs is not necessary. The RCP two-phase model in the Oconee RETRAN model has been revised, and that is a contributor to the improvement in the simulation.

6. Section 15.2.1.2, Transport Delay Model, is revised to state that this model is only turned off if reverse flow occurs in the primary loop piping.

Replace all text with the following:

*"The transport delay model is known to produce anomalous predictions if flow reversals occur. This model is turned off in the primary loop piping volumes if flow reversals are predicted in those volumes."*

Technical Justification: The RETRAN base model includes the enthalpy transport delay model in the primary loop piping volumes. Some steam line break analyses have situations in which flow reversals can occur in primary loop piping volumes. This is normally a result of idle reactor coolant pumps. It is knows that the transport delay model can predict anomalous results if flow reversals occur. The existing text reflected that this model was deleted in the steam line break analyses for this reason. The revision states that this model is deleted only when flow reversals are predicted, since all RCPs can remain in operation during some steam line break cases.

7. Section 15.3.1.1.4, Control Protection and Safeguards Systems, under "Feedwater Isolation" (Refer to Attachment 5, Item #40 in June 13, 2002 submittal), the text is revised to reflect the station as-built status for the AFIS modification

Change: *"The steam line break analyses do not credit automatic isolation of the main feedwater or emergency feedwater. Automatic isolation of feedwater is by the Automatic Feedwater Isolation System (AFIS) for Unit 1, and by the Main Steam Line Break Detection and Feedwater Isolation Circuitry for Units 2 and 3 (until AFIS is installed on those units)."*

To: *"The steam line break analyses do not credit automatic isolation of main feedwater or emergency feedwater by the Automatic Feedwater Isolation System (AFIS).*

Technical Justification: The AFIS System has now been installed on all three units. This editorial change is being made to keep the topical report consistent with current design.

8. Section 15.3.1.2.5, Critical Heat Flux Correlations, is revised to add the BWU-Z CHF correlation with an extended pressure range for the main steam line break with offsite power case for Mk-B11 fuel only.

Insert the following text at the end of Section 15.3.1.2.5:

*"The BWU-Z CHF correlation with the 0.98 Mk-B11 multiplier is also used for the with offsite power steam line break DNBR analysis for Mk-B11 fuel only. The minimum DNBR design limits are as follows for the following parameter ranges of applicability for this correlation and design limit:*

| | | Design DNBR |
|---|---|---|
| *Pressure (psia)* | *315 to 700* | *1.59* |
| | *700 to 1000* | *1.20* |
| | *1000 to 2465* | *1.19* |
| *Mass flux ($10^6$ lbm/hr-ft$^2$)* | *0.36 to 3.55* | |
| *Quality* | *less than 0.74"* | |

Technical Justification: The FANP Mk-B11 fuel assembly design has been licensed with the FANP BWU-Z CHF correlation with the 0.98 Mk-B11 multiplier. NRC approval of this correlation and the associated DNBR design limits is documented in Duke Power topical report "Thermal-Hydraulic Statistical Core Design Methodology," DPC-NE-2005P-A, Revision 3, Appendix D. The pressure range for this correlation in this NRC-approved topical report is from 400 to 2465 psia. Duke requests NRC approval to extend the low end of the pressure range from 400 psia to 315 psia. Duke has contracted with FANP to extend the pressure range for the BWU-Z correlation based on additional CHF test data that were not previously included in the Mk-B11 data base. The three additional test data points were tested at a test pressure of 315 psia. For all three of these data points, the measured power at the point of CHF exceeded the power predicted by the correlation, thus demonstrating the applicability of the correlation down to 315 psia. FANP has verified that the above DNBR limit (1.59 for pressures as low as 315 psia) remains valid for the above mass flux and quality ranges. The proposed extension of the lower end of the pressure range is only applicable to Mk-B11 fuel with the 0.98 Mk-B11 multiplier.

9. Section 15.3.2.1.4, Control Protection and Safeguards Systems, under "Feedwater Isolation" (Refer to Attachment 5, Item #41 in June 13, 2002 submittal), the text is revised to reflect the station as-built status for the AFIS modification

Change: *"The steam line break analyses do not credit automatic isolation of the main feedwater or emergency feedwater. Automatic isolation of feedwater is by the Automatic Feedwater Isolation System (AFIS) for Unit 1, and by the Main Steam Line Break Detection and Feedwater Isolation Circuitry for Units 2 and 3 (until AFIS is installed on those units)."*

To: *"The steam line break analyses do not credit automatic isolation of main feedwater or emergency feedwater by the Automatic Feedwater Isolation System (AFIS).*

Technical Justification: The AFIS System has now been installed on all three units. This editorial change is being made to keep the topical report consistent with current design.

<u>Chapter 16 - Small Steam Line Break Methodology Revisions</u>

10. Section 16.1.3, Boundary Conditions, is revised to include new modeling for the steam generator aspirator port.

Insert the following at the end of Section 16.1.3

*"<u>Aspirator Port Modeling During Downcomer Flooding (RETRAN-3D only)</u>*

*The small steam line break scenario, as modeled, can result in flooding of the steam generator downcomer due to excessive main feedwater flow. The RETRAN-3D prediction of aspirator flow appears anomalous when the downcomer overfills and floods the aspirator port. Continued steam flow into the aspirator is predicted, which is not physical. A control system is added that increases the aspirator junction loss coefficient as downcomer flooding is approached. This also forces the feedwater to flow up the entire length of the steam generator tube bundle, which maximizes heat transfer."*

Technical Justification: Review of RETRAN-3D cases showed anomalous aspirator flow into the steam generator downcomer when the downcomer was flooded. Flooding of the downcomer would be expected to terminate steam flow into the downcomer. This situation was addressed by adding a control system to increase the loss coefficient for the aspirator junction as the downcomer approached being flooded. This modeling change did not significantly change the overall results of the analysis, but was done for the purpose of addressing the observed anomaly.

11. Section 16.1.4, Physics Parameters, revise to make physics parameter values consistent with time in core life.

Doppler Coefficient

Change: *"The EOC least negative Doppler temperature coefficient value is assumed."*

To: *"The least negative Doppler temperature coefficient value is assumed consistent with the time in core life."*

Beta-Effective and Neutron Lifetime

Change: *"Thus, an EOC maximum value is assumed."*

To: *"Thus, a maximum value is assumed consistent with time in core life."*

Change: *"Therefore, typical EOC values are used."*

To: *"Therefore, values consistent with time in core life are used."*

Technical Justification: The original methodology uses end-of-cycle (EOC) values for the Doppler temperature coefficient, beta-effective, prompt neutron lifetime, delayed neutron fractions, and decay constants. In the proposed methodology, the values of these parameters are to be selected to be consistent with the time in core life. This is more typical of industry practice and is preferable relative to the arbitrary use of EOC values. It is noted that the small steam line break is not necessarily limiting at EOC. This revision will improve consistency in the methodology.

12. Section 16.1.5, Control Protection and Safeguards Systems, under "Feedwater Isolation" (Refer to Attachment 5, Item #44 in June 13, 2002 submittal), the text is revised to reflect the station as-built status for the AFIS modification

Change: *"The steam line break analyses do not credit automatic isolation of the main feedwater or emergency feedwater. Automatic isolation of feedwater is by the Automatic Feedwater Isolation System (AFIS) for Unit 1, and by the Main Steam Line Break Detection and Feedwater Isolation Circuitry for Units 2 and 3 (until AFIS is installed on those units)."*

To: *"The small steam line break analyses do not credit automatic isolation of main feedwater or emergency feedwater by the Automatic Feedwater Isolation System (AFIS).*

Technical Justification: The AFIS System has now been installed on all three units. This editorial change is being made to keep the topical report consistent with current design.