Duke Energy

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June 13, 2002

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, -3003, and 3005 In Support of Steam Generator Replacement

Enclosed herein, please find DPC-NE-3000-P, Revision 3, *"Thermal-Hydraulic Transient Analysis Methodology,"* DPC-NE-3003-P, Revision 1, *"Mass and Energy Release and Containment Response Methodology,"* and DPC-NE-3005-PA, Revision 2, *"UFSAR Chapter 15 Transient Analysis Methodology."* These reports are submitted for NRC review and acceptance as licensing Topical Reports under the NRC licensing topical report program for referencing in other license applications. Both the proprietary and non-proprietary versions of these Topical Reports are enclosed.

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

The replacement of steam generators at the Oconee Nuclear Station requires revision of three Duke Energy topical reports that detail the methodologies for analysis of transients and accidents. In addition, minor revisions to these reports are required periodically to maintain the technical content of these

APO1

reports current with the plant design and for consistency with other Duke topical reports. These reports and a summary of the revisions are as follows:

DPC-NE-3000-PA, Revision 2, *"Thermal-Hydraulic Transient Analysis Methodology"*, describes the RETRAN-02 system transient thermal-hydraulic analysis models, and the VIPRE-01 core thermal-hydraulic analysis models for analyzing non-LOCA transients and accidents for the Oconee, McGuire, and Catawba Nuclear Stations. Revision 2 was approved by the NRC in the Safety Evaluation Report (SER) enclosed with the letter dated October 14, 1998. Revision 3 includes the new RETRAN-3D model for the Oconee replacement steam generators. RETRAN-3D was reviewed and approved by the NRC by SER dated January 25, 2001. Compliance with the conditions and limitations in the NRC's RETRAN-3D SER are included in the attachments. The other revisions that are included in Revision 3 are minor technical or editorial changes necessary to maintain the topical report current since the last submittal in 1997. A description of each of the revisions and the associated technical justification are provided in the attachments. Attachments 1 and 2 are the proprietary and non-proprietary versions, respectively.

DPC-NE-3003-PA, *"Mass and Energy Release and Containment Response Methodology"*, describes the models for analyzing the mass and energy release from high-energy line breaks inside containment, and the resulting containment pressure and temperature response for the Oconee Nuclear Station. This report was approved by the NRC in the SER enclosed with the letter dated March 15, 1995. Revision 1 includes the new RELAP5 model for the replacement steam generator LOCA mass and energy release analyses, and the new RETRAN-3D model for the steam line break mass and energy release analyses. Revision 1 also includes a new GOTHIC 7.0 containment analysis model. Other minor revisions that are included in Revision 1 are minor technical or editorial changes necessary to maintain the topical report current since the last submittal in 1993. A description of each of the revisions and the associated technical justification are provided in the attachments. Attachments 3 and 4 are the proprietary and non-proprietary versions, respectively.

DPC-NE-3005-PA, Revision 1, *"UFSAR Chapter 15 Transient Analysis Methodology"*, describes the methodology for analyzing the UFSAR

Chapter 15 non-LOCA transients and accidents for the Oconee Nuclear Station. Revision 1 was approved by the NRC in the SER enclosed with the letter dated May 25, 1999. Revision 2 includes changes associated with the replacement steam generators and use of RETRAN-3D. Other minor revisions that are included in Revision 2 are minor technical or editorial changes necessary to maintain the topical report current since the last submittal in 1999. A description of each of the revisions and the associated technical justification are provided in the attachments. Attachments 5 and 6 are the proprietary and non-proprietary versions, respectively.

Approval of these topical report revisions is requested by June 3, 2003 to support startup of Oconee Unit 1 with the replacement steam generators currently scheduled for September, 2003.

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,

M. S. Tuckman

M. S. Tuckman
ATTACHMENTS

xc: with Non-Proprietary Attachments

L. A. Reyes, Regional Administrator

U.S. Nuclear Regulatory Commission, Region II
Atlanta Federal Center
61 Forsyth Street, SWW, Suite 23T85
Atlanta, GA 30303

M. C. Shannon, NRC Senior Resident Inspector (ONS)

xc: with Proprietary and Non-Proprietary Attachments
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AFFIDAVIT

1) I am Executive 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 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.

M. S. Tuckman

M. S. Tuckman

(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 June 13, 2002; Subject: Revisions to Topical Reports DPC-NE-3000, -3003, and 3005 in Support of Steam Generator Replacement. 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.

i) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

ii) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

iii) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

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.

M. S. Tuckman

M. S. Tuckman

(Continued)

M. S. Tuckman 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.

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to on this 13TH day of

June, 2002

Mary P. Nelms
Notary Public

My Commission Expires:

JAN 22, 2006

SEAL

Attachment 2

Thermal-Hydraulic Transient Analysis Methodology

DPC-NE-3000 (Non-Proprietary)
Revision 3

June 2002

Nuclear Engineering Division
Nuclear Generation Department
Duke Power Company

DPC-NE-3000 Non-Proprietary, Revision 3

Description and Technical Justification

Revision 3 to DPC-NE-3000 consists of changes necessary to model the replacement steam generators with RETRAN-3D, minor technical and editorial changes to update the report since Revision 2 was submitted in 1997, and a statement of the compliance with the NRC's SER on RETRAN-3D. Each of the revisions is described in detail, and model changes are supported by technical justification. The RETRAN-3D modeling has been reviewed by Computer Simulation & Analysis (CSA), Inc., the developer of the RETRAN-3D code. There are currently no other organizations using RETRAN-3D for modeling B&W-designed reactors, so a peer review other than by the code vendor was not possible.

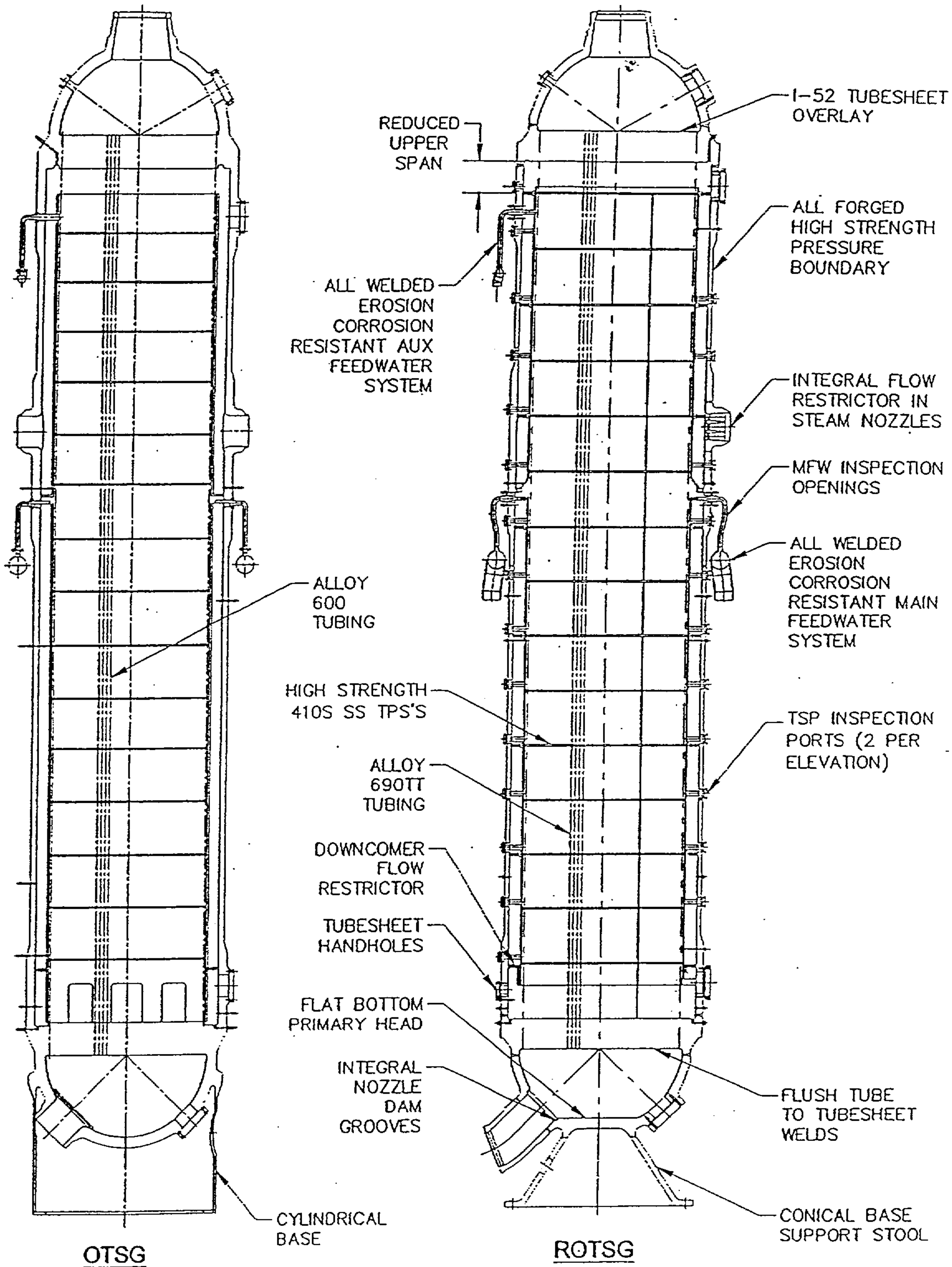
The Oconee replacement steam generators (ROTSGs) are being manufactured in Canada by Babcock & Wilcox Canada (BWC). The ROTSGs are quite similar functionally to the original OTSGs (Figure 1), and are characterized as a like-for-like replacement. The design differences that are important for thermal-hydraulic modeling are 1) flow-restricting orifices in the steam outlet nozzles, 2) Inconel-690 tubes, 3) 15,631 vs. 15,531 tubes, 4) thinner pressure vessel / wider downcomer, 5) thinner tubesheets resulting in 3.625 inch longer heated tube length, 6) 1.2% greater heat transfer area, and 7) more water in the steam generator. Of these design changes only the flow-restricting steam exit nozzles have a significant impact on the UFSAR transient and accident analyses. These nozzles reduce the effective size of steam line breaks during blowdown of a steam generator to 1.804 ft^2. The other design changes will not cause a significant change in plant response during transients and accidents. For that reason the modeling of the ROTSGs is similar to the modeling of the OTSGs, with changes limited to those described herein.

The system thermal-hydraulic transient analysis methodology through Revision 2 was based on the Electric Power Research Institute's RETRAN-02 code. The next generation in the RETRAN family of codes, RETRAN-3D, was approved by the NRC in the SER dated January 25, 2001. Revision 3 to DPC-NE-3000 describes the modeling for Oconee with RETRAN-3D replacing RETRAN-02. [

] All of the details of the transition to using RETRAN-3D instead of RETRAN-02 for Oconee modeling are presented. The conditions and limitations in the NRC's SER for RETRAN-3D are also addressed.

The minor technical and editorial revisions are not necessarily associated with either the ROTSGs or the transition to RETRAN-3D. These changes are necessary to maintain the content of the report consistent with the current plant design and modeling, as well as to correct errors. No technical justification is necessary for editorial revisions and corrections. The changes are presented in the order that they appear in DPC-NE-3000. New Appendix B describes the Oconee ROTSG modeling. New Appendix C addresses the limitations and conditions in the RETRAN-3D SER as relates to the application to Oconee.

Figure 1 OTSG / ROTSG Comparison



Changes and Technical Justification

Cover Page and Frontal Pages

1. The revision and date will be revised

2. The table of contents and the lists of tables, figures , and acronyms will be updated

Chapter 1

3. New Section 1.3 "RETRAN-3D Code Description" (editorial)

    1.2 RETRAN-3D Code Description

    RETRAN-3D (Reference 1-18) was developed by Computer Simulation & Analysis, Inc. for EPRI to enhance and extend the simulation capabilities of the RETRAN-02 code. RETRAN-3D has many new and enhanced capabilities relative to RETRAN-02, in particular, a 3-D kinetics core model, improved two-phase models, an improved heat transfer correlation package, and an implicit numerical solution method. Most of the capabilities of the RETRAN-02 code have been retained within RETRAN-3D as options, except for a limited number of models and correlations that were not in use. The NRC SER for RETRAN-3D is dated January 25, 2001 (Reference 1-19). The SER includes limitations and conditions on the use of the code for licensing applications.

4. New Section 1.7 "Interface with Duke Reload Design Methodology Topical Reports" (editorial)

    1.7 Interface with Duke Reload Design Methodology Topical Reports

    This report is referenced by DPC-NE-2009-PA, "Duke Power Company Westinghouse Fuel Transition Report." (Reference 1-20). Section 6.1 of DPC-NE-2009 lists the content of DPC-NE-3000 that was affected by modeling Westinghouse RFA fuel. Since DPC-NE-2009 was reviewed and approved by the NRC, the content of Section 6.1 is inserted into DPC-NE-3000 with Revision 3.

5. New Section 1.8 "Appendices" (editorial)

    1.8 Appendices

    Appendix A was added in Revision 2 to describe the Mk-B11 fuel assembly used at Oconee and how it will be simulated with the RETRAN-02 and VIPRE-01 models. This modeling of the Mk-B11 fuel assembly is also applicable to RETRAN-3D.

    Appendix B was added in Revision 3 to describe the Oconee replacement steam generator (ROTSG) modeling with RETRAN-3D.

    Appendix C was added in Revision 3 to address the RETRAN-3D SER conditions and limitations as relates to the modeling for Oconee.

6. Section 1.8: New references added (editorial)

    1-18 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI, NP-7450(A), Volumes 1-4, Revision 5, July 2001

    1-19 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

    1-20 Duke Power Company Westinghouse Fuel Transition Report, DPC-NE-2009P-A, Duke Power, December 1999

7. Renumbered sections (editorial)

Chapter 2

8. Section 2.1.5.3, Emergency Feedwater System, second paragraph, requires a correction on the start signals for the EFW pumps. (error correction)

    Change: "All three pumps receive a start signal following either of two indications that MFW is lost - low MFW pump hydraulic oil pressure or low steam generator level."

    To: "All three pumps receive a start signal following a loss of MFW pumps as indicated by low MFW pump turbine hydraulic control oil pressure. The two motor-driven EFW pumps also start on low steam generator level."

9. Section 2.1.5.4, Steam Line Break Detection and Mitigation Circuitry, needs to be revised to reflect a station design change. This design feature is being replaced by the Automatic Feedwater Isolation System (AFIS). Since Unit 1 is the only unit with AFIS installed, both designs will be discussed. Replace the title of Section 2.1.5.3 and revise and expand the current text to include AFIS. (design description)

    2.1.5.3 MFW and EFW Automatic Isolation

    The Steam Line Break Detection and Mitigation Circuitry uses low steam generator outlet pressure and turbine header pressure as the initiating parameters for automatic MFW isolation. When a steam line break is sensed by steam generator outlet pressure falling below the setpoint, both trains of MFW control valves, MFW block valves, startup feedwater control valves and startup feedwater block valves are closed. Additionally, both MFW pumps are tripped. Tripping both MFW pumps will trip the reactor, trip the turbine and start the motor-driven EFW pumps. The circuitry also stops the turbine-driven EFW pump. The motor-driven EFW pumps are not automatically isolated. This design is being replaced by the Automatic Feedwater Isolation System (AFIS) design.

    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.

10. Section 2.1.6, Dissimilarities Between Units, fourth paragraph, needs to be revised to update to the current fuel assembly design, which are designated as Mk-B10 and Mk-B11 fuel. (design description)

    Change: "For example, Oconee now uses Mark BZ fuel assemblies, which incorporate Zircaloy spacer grids for enhanced neutron economy. However, some Mark B assemblies with the smaller Inconel spacer grids remain in the cores, since only one-third of the assemblies are replaced during each reload.

    To: "Oconee now uses Mk-B10 and Mk-B11 fuel assemblies, with Mk-B11 currently being loaded. Refer to Appendix A for a description of Mk-B11 fuel assemblies."

11. Section 2.2.2.2, Steam Generators, fifth paragraph, needs to be revised to include [ ] (model revision)

    Change: "The steam outlet annulus, [ ]

    To: "The steam outlet annulus is usually [ ]

    Technical Justification: Recent Oconee RETRAN modeling experience has required accurate simulation of the temperature of the steam generator shell. This is necessary due to the operational limit for OTSGs referred to as the "tube-to-shell ΔT limit". To accurately calculate the transient value of the SG shell temperature, the RETRAN model [ ] This detail was not included in the previous model since it was not recognized as being necessary.

12. Section 2.2.6.1, Power Generation, first paragraph, needs to be revised to reflect the use of the 1979 ANS decay heat standard. (model revision)

    Change: "Post-trip decay heat energy is calculated with a model of eleven delayed gamma emitters, plus a contribution for heavy element (U-239 and Pu-239) decay. The resulting decay heat is a close fit to the proposed 1971 ANS Standard (Reference 2-1)."

    To: Post-trip decay heat is calculated with the built-in 1979 ANS Standard (Reference 2-2) decay heat option. Input data for the decay heat model is selected consistent with the application.

    Also, delete "(Reference 2-2)" in the second paragraph.

    Technical Justification: The 1979 ANS Standard decay heat model is widely used and accepted in the industry. The previous use of the 1971 ANS Standard resulted from the 1979 standard not yet being available in the RETRAN-02 code.

13. Section 2.2.6.7, Local Conditions Heat Transfer, add a new paragraph to include use of the local conditions model in the [ ] (model revision)

Change: Insert the following new paragraph at the end of Section 2.2.6.7

["The local conditions heat transfer option is used to model the wall heat transfer in the] This model gives a more accurate calculation of heat transfer in the presence of a mixture level and can be important for some applications. This modeling approach may also be used for other passive structural conductors when additional accuracy is needed."

Technical Justification: The use of the local conditions heat transfer option in the [ ] This detail was not in the previous model since it was not recognized as being necessary. This modeling approach is consistent with the intent of the local conditions heat transfer option.

14. Section 2.2.7.8, Volume Flow Calculation, correct an incorrect statement as to the basis for the selection of this code option. (error correction)

Change: "This choice is unimportant since the vector-momentum model is not used."

To: "The choice of the donor cell option vs. the arithmetic average option is not important for the intended applications."

Technical Justification: The statement that the choice of the donor cell option vs. the arithmetic average option for calculation of the average volume flow for the momentum flux pressure change was incorrectly associated with the vector momentum model. This option choice is applicable whenever the momentum flux terms are active in the RETRAN momentum equation, which is the default equation used for nearly all junctions. The revision more correctly indicates that the donor cell option is selected, and that this is not important for the intended applications. Either option is an acceptable choice.

15. Table 2.2.1, Oconee Base Model Heat Conductors, needs revision to indicate that the conductor modeling for the [ ] (model revision)

Change: Add the following note after the [ ] in Table 2.2.1 at the bottom of page 2-61.

"Note: [ ]"

Technical Justification: [

] This detail was not in the previous model since it was not recognized as necessary.

16. Section 2.4, References, delete Reference 2-1 since no longer applicable (editorial)

<u>Chapter 3</u>

17. Section 3.1.2.1, Reactor Core, include revisions to update to the two fuel assembly designs currently in use, the Framatome Advanced Nuclear Products Mk-BW and the Westinghouse Robust Fuel Assembly. (design description)

    Change: "Each fuel rod contains stacked $U0_2$ fuel pellets surrounded by Zircaloy-4 cladding, with a small gap between the pellets and the cladding. The Zircaloy guide thimbles provide a channel for control rod insertion."

    To: Two fuel assembly designs are currently in use. The Framatome Advanced Nuclear Products Mk-BW fuel assembly, and the Westinghouse Robust Fuel Assembly (RFA). The Mk-BW fuel rod contains stacked UO_2 fuel pellets inside Zircaloy-4 cladding, with a small gap initially between the pellets and the cladding. The Mk-BW design has Zircaloy-4 guide thimbles to provide a channel for control rod insertion. The RFA fuel assembly design is similar, but uses ZIRLO™ for the cladding and guide thimble material.

18. Figure 3.1-2 Replace illustration of Westinghouse Optimized Fuel Assembly (OFA) with Westinghouse Robust Fuel Assembly (RFA). The OFA design is no longer used. (design description)

    Change: Replace illustration of OFA design with illustration of RFA design.

19. Section 3.2.6.1, Power Generation, first paragraph, needs to be revised to reflect the use of the 1979 ANS decay heat standard. (model revision)

    Change: "Post-trip decay heat energy is calculated with a model of eleven delayed gamma emitters, plus a contribution for heavy element (U-239 and Pu-239) decay. The resulting decay heat is a close fit to the proposed 1971 ANS Standard (Reference 3-1)."

    To: Post-trip decay heat is calculated with the built-in 1979 ANS Standard (Reference 3-2) decay heat option. Input data for the decay heat model is selected consistent with the application.

    Also, delete "(Reference 3-2)" in the second paragraph.

    Technical Justification: The 1979 ANS Standard decay heat model is widely used and accepted in the industry. The previous use of the 1971 ANS Standard resulted from the 1979 standard not yet being available in the RETRAN-02 code.

20. Section 3.2.6.7, Local Conditions Heat Transfer, add a new paragraph to include use of the local conditions model[ ] (model revision)

    Change: Insert the following new paragraph at the end of Section 3.2.6.7

    "The local conditions heat transfer option is also used to model the[

    ]

    Technical Justification: The use of the local conditions heat transfer option in the [ ] in Section 3.2.3.3 and is only being added here for completeness.

21. Section 3.2.7.8, Volume Flow Calculation, correct an incorrect statement as to the basis for the selection of this code option. (error correction)

    Change: "This choice is unimportant since the vector-momentum model is not used."

    To: "The choice of the donor cell option vs. the arithmetic average option is not important for the intended applications."

    Technical Justification: The statement that the choice of the donor cell option vs. the arithmetic average option for calculation of the volume momentum flux term was incorrectly associated with the vector momentum model. This option choice is applicable whenever the momentum flux terms are active in the RETRAN momentum equation, which is the default equation used for nearly all junctions. The revision more correctly indicates that the donor cell option is selected, and that this is not important for the intended applications. Either option is an acceptable choice.

22. Section 3.4, References, delete Reference 3-1 since it is no longer applicable. Revise Reference 3-11 to the latest revision and date. (editorial)

    "3-11 The BWU Critical Heat Flux Correlations, BAW-10199P-A, Framatome Advanced Nuclear Products, August 1996"

<u>Chapter 6</u>

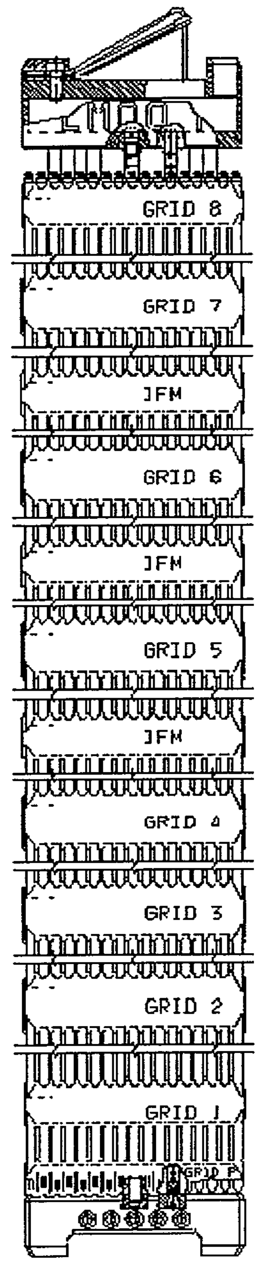
23. Add the following paragraphs at the end of Chapter 6 to summarize the content of the appendices. (editorial)

    "Appendix A was added in Revision 2 to describe the Framatome Advanced Nuclear Products Mk-B11 fuel assembly with M5 cladding in use at Oconee. Use of the BWU-Z CHF correlation applicable to the Mk-B11 design was also included.

    Appendix B was added in Revision 3 to describe the modeling of the Oconee replacement steam generators with RETRAN-3D.

    Appendix C was added in Revision 3 to address the conditions and limitations of the RETRAN-3D SER on the application of RETRAN-3D for Oconee transient simulation modeling."

Figure 3.1-2 Westinghouse Robust Fuel Assembly



APPENDIX B

METHODOLOGY REVISION FOR
OCONEE REPLACEMENT STEAM GENERATORS

ROTSG Design Description

This appendix describes the Oconee RETRAN model revisions necessary for simulation of the replacement once-through steam generators (ROTSGs) being manufactured by Babcock & Wilcox Canada (BWC). The first pair of ROTSGs are scheduled for installation in Oconee Unit 1 in 2003. The other two Oconee units are scheduled for replacement outages in 2004. The design of the ROTSGs enables what is characterized as a "like-for-like" replacement due to the close similarity in the performance of the component. Due to the similarity of the ROTSGs with respect to the original OTSGs, the RETRAN modeling is also maintained very similar. The new RETRAN models for the ROTSGs will be inserted into the Oconee RETRAN model described in Chapter 2 of this report. The significant design differences are as follows. Figure B-1 shows a comparison of the OTSG and ROTSG designs.

1) Flow-restricting orifices in the steam outlet nozzles
2) Inconel-690 tubes
3) 15,631 vs. 15,531 tubes
4) Thinner pressure vessel / wider downcomer
5) Thinner tubesheets resulting in 3.625 inch longer heated tube length
6) 1.2% greater heat transfer area
7) More water in the steam generator.

Other than the ROTSG dimensional design data, Duke is also using steady-state thermal-hydraulic data provided by BWC as the reference data for the ROTSG initial conditions. These data included the pressure distribution, void fraction distribution, steam superheat profile, and water masses. The BWC simulations are three-dimensional, whereas RETRAN predictions are one-dimensional, so when comparing the results of the two codes it must recognized that there is some approximation.

RETRAN-3D Code

The Electric Power Research Institute's (EPRI) RETRAN-3D code (Reference B-1) is used for the Oconee RETRAN Model with ROTSGs. Previous modeling described in the body of this report with the original OTSGs used the EPRI RETRAN-02 code. RETRAN-3D was approved by the NRC in the SER dated January 25, 2001 (Reference B-2). The most significant change in the Oconee RETRAN methodology resulting from this RETRAN code version change is in the two-phase modeling of the secondary side of the steam generators. With RETRAN-02 the secondary side is modeled using [

] All of the details of the transition to using RETRAN-3D instead of RETRAN-02 for Oconee modeling are presented. It is noted that the three-dimensional core model capability in RETRAN-3D is not included in this methodology, as well as some of the other new models. The conditions and limitations in the NRC's SER for RETRAN-3D are addressed in Appendix C.

ROTSG Nodalization

The ROTSG nodalization is shown in Figure B-2. This nodalization can be compared to the OTSG nodalization in Figure 2.2-1. The only significant difference is that the [

]

Details of RETRAN-3D Modeling

The selection of RETRAN-3D models, code options, and other input specifications for the Oconee RETRAN Model with ROTSGs is presented and justified. The sequence of the presentation that follows is consistent with the RETRAN-3D input file card sequence in Volume 3 Revision 5 of the EPRI RETRAN-3D code documentation. Volume 3 Revision 6 of the EPRI RETRAN-02 code documentation is used as the point of reference for RETRAN-02 code models, options, and input. Only those models, options, and input that differ from RETRAN-02 modeling as described in the body of this report are presented. The RETRAN-3D models that replace RETRAN-02 models (i.e., the RETRAN-02 models are no longer available to the user - such as the heat transfer correlation package and the numerical solution method), and have been reviewed and approved by the NRC during the generic review of RETRAN-3D, are not presented. ROTSG design dimensional data input to the RETRAN-3D code, such as the length of the ROTSG tubes or the thickness of the shell, are not considered methodology, and similar to the body of this report are not included.

Card 01000Y - Problem Control and Description Data

[ ]

W30-I IHTMAP - (heat transfer map option flag) = 2 - Custom modification to allow condensation heat transfer with the forced convection heat transfer map.

Application: RETRAN-3D analyses for ROTSGs use this option.

Technical Justification: The NRC-approved RETRAN-02 methodology for Oconee uses the forced convection heat transfer map with a custom modification to allow use of condensation correlations when appropriate. A similar custom modification has been made to RETRAN-3D to allow access to the condensation correlations in combination with the forced convection heat transfer correlations set by setting IHTMAP=2. The condensation correlations are used whenever a conductor surface temperature is below the saturation temperature and steam is present.

W38-I JFLAG - (average volume flow calculation option) = 0 - arithmetic average

Application: This option is applicable to all junctions except those with momentum flux turned off.

Technical Justification: The arithmetic average option for the calculation of the average volume flow is the vendor recommendation.

<u>Card 08XXXY - Junction Data</u>

[ ]

W17-I IFRJ - (slip flow regime/orientation flag) = (any value 0-9) - vertical flow path
= -99 - no slip

Application: For the [ ], this flag is set to a value of 0-9 since the flowpaths are vertically oriented. For all other junctions where two-phase conditions may develop this flag is set to -99 to deactivate the algebraic slip model. These junctions will then have equal liquid and vapor velocities if two-phase conditions develop.

Technical Justification: See Card 01000Y above for discussion regarding use of the [ ]. This flag is necessary to control which junctions use this model, and also to designate the vertical orientation of these junctions in the ROTSG secondary.

W21-R ANGLJ1 - (angle of junction relative to the "from" volume)
W22 -R ANGLJ2 - (angle of junction relative to the "to" volume)

Application: The following junctions have been assigned the following angles to correctly adjust the momentum flux terms.

Technical Justification: Vendor recommendations for modeling the indicated junction angles are the basis for the selected values.

Card 15XXXY - Conductor Data

W5-I IMCL (indicator for selecting heat transfer correlations at left surface) = 41 or 48. Custom modification to allow the Dittus-Boelter forced convection heat transfer correlation to either liquid or steam to be specified, regardless of the local property conditions adjacent to the left conductor wall

Application: IMCL = 41 will select the Dittus-Boelter forced convection to liquid heat transfer correlation. IMCL = 48 will select the Dittus-Boelter forced convection to vapor heat transfer correlation. These are custom coding changes used to select one of these correlations regardless of the local fluid conditions adjacent to a conductor.

References

B-1 RETRAN-3D- A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI, NP-7450(A), Volumes 1-4, Revision 5, July 2001

B-2 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

Figure B-1

OTSG / ROTSG Comparison

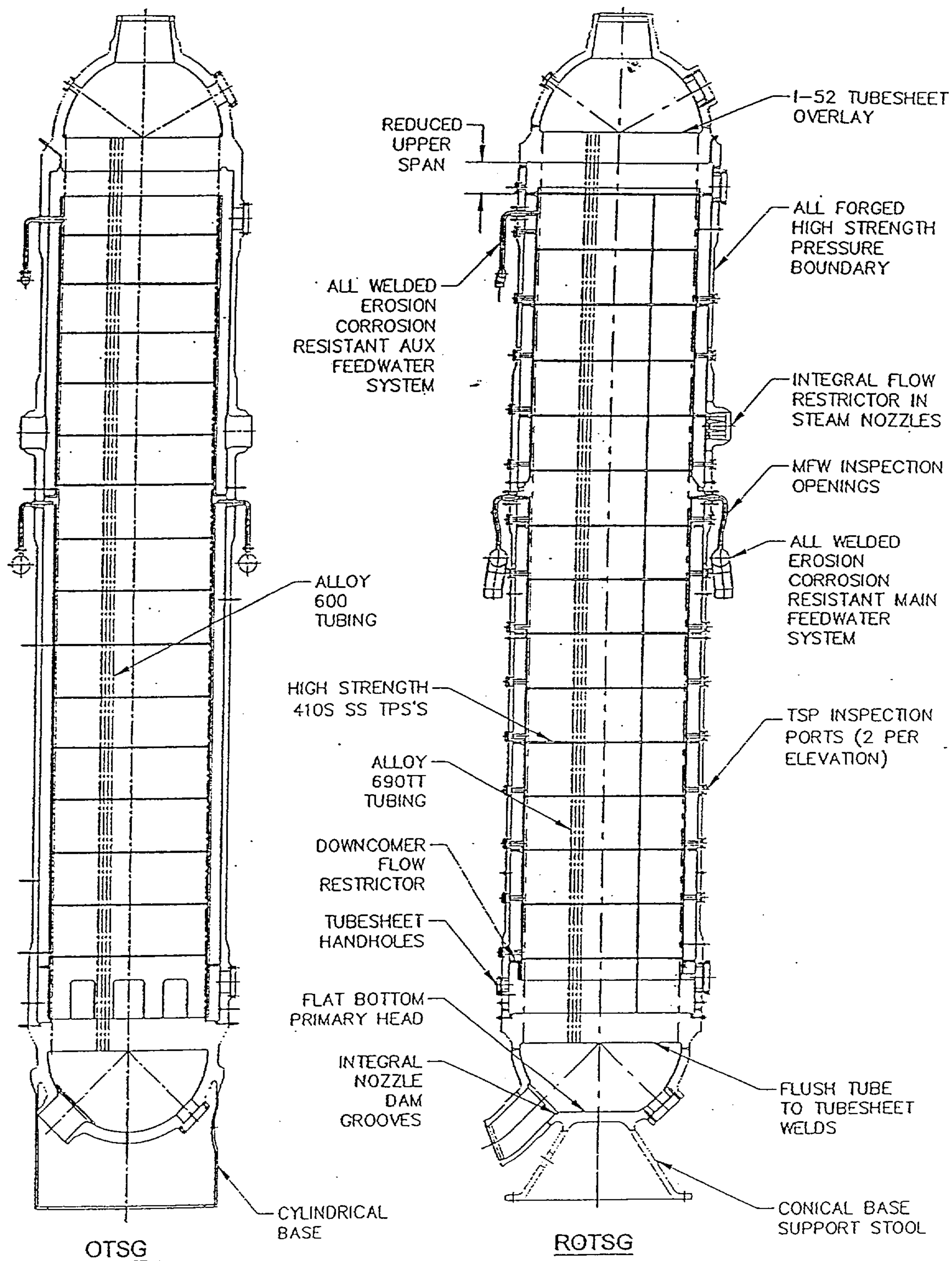


Figure B-2

ROTSG Nodalization

Figure B-3

ROTSG Void Fraction Comparison

APPENDIX C

EVALUATION OF RETRAN-3D SER CONDITIONS AND LIMITATIONS FOR THE OCONEE RETRAN MODEL WITH ROTSGs

Purpose

This appendix evaluates the conditions and limitations in the RETRAN-3D SER (Reference C-1) for the application of RETRAN-3D to the Oconee Nuclear Station with replacement steam generators (ROTSGs). The results of this evaluation demonstrate that the use of the RETRAN-3D code for this application, as described in Appendix B, has been appropriately justified and is within the SER conditions and limitations. Therefore, the approval for use of RETRAN-3D as stated in the SER for the scope of approval specified in the SER can be credited. The Duke version of the RETRAN-3D code actually used is described first, including details on the custom code modifications that have been incorporated.

Description of the RETRAN-3D Code Version Used and Duke Code Modifications

RETRAN-3D MOD003.1DKE is the current Duke Power version of the standard Electric Power Research Institute (EPRI) RETRAN-3D MOD003.1 code, the NRC-approved code. This EPRI version includes the revisions agreed to during the SER review process. The Duke Power version, designated by the suffix "DKE", consists of two types of revisions to the standard EPRI version. The first type of revision is code error corrections. Duke periodically updates the code version in use to include error corrections obtained from Computer Simulation & Analysis, Inc. (CSA), the EPRI contractor for the RETRAN-3D code. The second type of revision is Duke custom code modifications purchased from CSA to address Duke-specific modeling needs. Each of these custom code changes are described in detail for NRC review and approval. The error corrections are not presented. All error corrections and code modifications are developed and controlled under CSA's Appendix B QA program. All Duke RETRAN-3D versions used for safety-related applications are certified and controlled under Duke's Appendix B software quality assurance program (Reference C-2). Any future modifications to the RETRAN-3D code versions used by Duke for safety-related applications that constitute significant model revisions or new models will be submitted for NRC review and approval. Code modifications that consist of error corrections or user features will be implemented under QA processes, but will not be submitted for NRC review.

Duke Code Modification #1 — Allow Access to the Condensation Heat Transfer Correlations With the Forced Convection Heat Transfer Map

Initialization of the ROTSGs using RETRAN-3D uses the forced convection heat transfer map. This is selected by setting variable IHTMAP on the 01000Y card to a value of zero. This standard forced convection option does not allow access to the condensation heat transfer correlations in RETRAN-3D. A code modification was implemented to allow access to the

condensation heat transfer correlations by setting IHTMAP to a value of 2. This option gives the forced heat transfer map, but allows condensation heat transfer to be modeled when appropriate for the local conditions present. This can be important in several transient conditions, such as when the primary water flowing through the steam generator tubes cools to below the secondary saturation temperature. In this situation an appropriate condensation heat transfer coefficient will be selected from the heat transfer correlation set. The technical justification for this code modification is that it allows a correct heat transfer correlation to be used for situations when condensation heat transfer occurs. This modification is functionally equivalent to a similar update made to RETRAN-02, which has been reviewed and approved for use by the NRC.

<u>Duke Code Modification #2</u> Allow the User to Specify the Dittus-Boelter Heat Transfer Correlation for a Specific Conductor

The RETRAN-3D heat transfer correlation package selects an appropriate heat transfer correlation for each conductor surface based on fluid conditions in adjacent volumes. Under some conditions it is useful to be able to select a specific correlation for a given conductor surface rather than using the code-selected correlation. This code modification allows the user to specify either the Dittus-Boelter liquid or vapor correlation for the left surface of a particular conductor. This will then override the code-selected correlation. Word IMCL on Card 15XXXY is set to a value of 41 to specify the liquid correlation, and to a value of 48 to specify the vapor correlation at the left surface of a conductor. The need for this modeling capability arose during analyses of the ROTSG upper shell heat transfer following a steam line break. Due to water carryout into the steam outlet annulus (the volume adjacent to the upper shell), the heat transfer was potentially too high for an analysis in which less heat transfer was conservative. This code modification allowed specifying use of the Dittus-Boelter correlation for vapor to conservatively model the heat transfer to the shell conductor. The code modification also allows specification of the Dittus-Boelter correlation for liquid as another modeling option. The technical justification for this code modification is that a capability to specify a heat transfer correlation for specific applications is appropriate.

<u>Duke Code Modification #3</u>

Duke Code Modification #4

Duke Code Modification #5

Evaluation of RETRAN-3D SER Conditions and Limitations

1. Multidimensional neutronic space-time effects cannot be simulated as the maximum number of dimensions is one. Conservative usage has to be demonstrated.

Staff Position: RETRAN-3D has been modified to include a 3-dimensional nodal kinetics model based on the analytic nodalization method similar to accepted codes. The code has been assessed by calculation of the response of the SPERT prompt-critical tests and has been confirmed by the staff by comparisons with calculations performed with the NESTLE and TORT codes. The staff concludes that the code can adequately predict the response to prompt-critical events such as the PWR rod ejection accident and the BWR rod drop accident. If void generation occurs from an initially un-voided case, the user will have to justify crediting this negative feedback in the analysis.

The code was used by a participant in the Nuclear Energy Agency's International Standard Problem calculation of a hypothetical main steam line break (MSLB) at the Three Mile Island Unit 1 plant. The results of the calculation comparison indicates that RETRAN-3D is comparable to any of the other participating codes.

RETRAN-3D is approved for main steam line break analyses subject to the following conditions. Thermal-hydraulic effects can have a large impact on the cross section evaluation and thus on the resulting power distribution and magnitude. Therefore, the licensee must justify the primary side nodalization for mixing in the vessel and core. The licensee must also evaluate the uncertainties in the modeling.

Duke Position: The RETRAN-3D three-dimensional kinetics model is not used.

2. There is no source term in the neutronics and the maximum number of energy groups is two. The space-time options assume an initially critical system. Initial conditions with zero fission power cannot be simulated by the kinetics. The neutronic models should not be started from subcritical or with zero fission power without further justification.

Staff Position: The basic models in RETPAN-3D are unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not analyze from subcritical or zero fission power initial conditions.

3. A boron transport model is unavailable. User input models will have to be reviewed on an individual basis.

Staff Position: As noted previously in this report, boron transport is handled as a "contaminant" by the "general transport model." This model uses first order accurate upwind difference scheme with an implicit temporal differencing. This approach is well known for being highly diffusive, especially if the Courant limit is exceeded. Since RETRAN-3D has the same model as RETRAN-02 MODOO3 and subsequent versions that have been approved for use, the RETRAN-3D model is also approved with the caveat that the potential to produce misleading results with this scheme necessitates careful review of the results for any case where boron transport/dilution is important.

Duke Position: Previously approved in RETRAN-02 and for Duke applications using RETRAN-02.

4. Moving control rod banks are assumed to travel together The BWR plant qualification work shows that this is an acceptable approximation.

Staff Position: The control bank limitation is applied only to the one-dimensional kinetics model. The staff agrees that the 3-dimensional kinetics model need not be restricted in this way.

Duke Position: Resolved per the Staff Position

5. The metal-water heat generation model is for slab geometry The reaction rate is therefore under-predicted for cylindrical cladding. Justification will have to be provided for specific analyses.

Staff Position: The basic models in RETRAN-3D are unchanged and, therefore, this condition of use applies. However, since RETRAN-3D is not being reviewed for loss-of-coolant accident analysis, where core uncovery and heatup are significant, this condition does not occur in the transients for which application of RETRAN-3D has been reviewed.

Duke Position: Duke does not use the metal-water heat generation model.

6. Equilibrium thermodynamics is assumed for the thermal-hydraulics field equations although there are nonequilibrium models for the pressurizer and the subcooled boiling region.

Staff Position: The RETRAN-3D five equation model permits thermal-hydraulic nonequilibrium between the liquid and vapor phases. While it allows subcooled liquid and saturated steam to be concurrently present, it does not account for subcooled liquid and superheated vapor being concurrently present. Use of the code is not approved for LOCA. Also, the user must be aware of this limitation and avoid conditions which will place subcooled liquid and superheated vapor in contact.

Duke Position: Duke does not use the RETRAN-3D five equation model.

7. While the vector momentum model allows the simulation of some vector momentum flux effects in complex geometry the thermal-hydraulics are basically one-dimensional.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this comment still applies.

Duke Position: As described in Appendix B, Duke is proposing to use the vector momentum option with junction angle input for certain junctions where the momentum flux terms are considered to be potentially important. It is acknowledged that the thermal-hydraulics are basically one-dimensional. Duke's use of this model is based on vendor recommendations.

8. Further justification is required for the use of the homogeneous slip options with BWRs.

Staff Position: RETRAN-3D has five slip equation options for the user to choose from, three of which are retained from RETRAN-02 for compatibility. The recommended model options are based on the Chexal-Lellouche drift flux correlation. The first is the algebraic slip model, which

is approved for use with BWR bundle geometry as given in condition (9). The second is a form of the dynamic slip model that uses the Chexal-Lellouche drift flux correlation to evaluate the interfacial friction approved in condition (10). The user must justify the use of any other slip options.

Duke Position: Duke is not modeling BWRs.

9. The drift flux correlation used was originally calibrated to BWR situations and the qualification work for both this option and for the dynamic slip option only cover BWRs. The drift flux option can be approved for BWR bundle geometry if the conditions of (16) are met.

Staff Position: The Chexal-Lellouche drift flux model has been used in comparisons with FRIGG-2 and FRIGG-4 void fraction data and is acceptable for use in BWR bundle geometry.

Duke Position: Duke is not modeling BWRs.

10. The profile effect on the interphase drag (among all the profile effects) is neglected in the dynamic slip option. Form loss is also neglected for the slip velocity. For the acceptability of these approximations refer to (17).

Staff Position: Form loss terms have been included in the RETRAN-3D dynamic slip model. The Taugl form of the dynamic slip equation also includes profile effects in the interphase drag model. These RETRAN-3D model improvements adequately address the concerns and the model is approved for use when the Chexal-Lellouche model is used to compute the interphase friction. Approval is subject to the conditions given in (16) for the Chexal-Lellouche drift flux correlation. Users must justify use of any other dynamic slip option.

Duke Position: Duke is not using the dynamic slip option.

11. Only one-dimensional heat conduction is modeled. The use of the optional gap linear thermal expansion model requires further justification.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use still applies.

Duke Position: Duke is not using the optional linear gap thermal expansion model.

12. Air is assumed to be an ideal gas with a constant specific heat representative of that at containment conditions. It is restricted to separated and single-phase vapor volumes. There are no other noncondensables.

Staff Position: RETRAN-3D has been extended to include a general noncondensable gas capability which resolves the original concern. However, the noncondensable gas flow model is approved for use subject to the following restriction.

As noted in Section 111.3.0 of the RETRAN-3D Theory Manual (Reference 4), none of the models available for calculating critical flow are appropriate when noncondensable gases are present. Consequently, the code automatically bypasses the critical flow model when noncondensable gases are present in a junction. Users must confirm that noncondensable flows do not exceed appropriate critical flow values or justify use of values that may exceed critical flow values.

Duke Position: Duke is not using the noncondensable gas flow model.

13. The use of the water properties polynomials should be restricted to the subcritical region. Further justification is required for other regions.

Staff Position: For enthalpies less than approximately 820 Btu/lbm, the difference between the ASME and RETRAN-3D curve fit values of the specific volume range from less than 0.2 percent to approximately 1.3 percent for pressures ranging from 0.1 to 6,000 psia. Further, for enthalpies greater than 820 Btu/lbm and pressures greater than 4200 psia, the differences in specific volume are also less than 1.0 percent. RETRAN-3D is approved for use with PWR ATWS analyses where the peak pressure resides in the regions described above.

For enthalpies greater than 820 Btu/lbm and pressures between 3200 and 4200 psia, the differences in specific volume increase as the enthalpy increases and the pressure decreases. The maximum error of approximately 3.8 percent occurs at the critical point. PWR ATWS analysis using RETRAN-3D in this region will require additional justification that the difference in specific volume does not adversely affect the calculation of the peak pressure.

Duke Position: Duke will address the above condition if an application encounters conditions in the region of concern.

14. A number of regime-dependent minimum and maximum heat fluxes are hardwired. The use of the heat transfer correlations should be restricted to situations where the pre-CHF heat transfer or single-phase heat transfer dominates.

Staff Position: RETRAN-3D contains both the "forced convection option" contained in RETRAN-02 which is the basis for this restriction, and a second option referred to as the "combination heat transfer map." If the first option is chosen, the "forced convection option," approval is granted only for use in pre-CHF and single-phase heat transfer regimes. If the second option is chosen, the "combination heat transfer map," then there are no discontinuities between successive heat transfer regimes and the appropriate heat transfer value should result. Therefore, the combination heat transfer option is approved for use.

Duke Position:

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15. The Bennett flow map should only be used for vertical flow within the conditions of the data base and the Beattie two-phase multiplier option requires qualification work

Staff Position: The Beattie two-phase multiplier has been removed from RETRAN-3D. The Govier horizontal flow map has been added to supplement the Bennett map for vertical flow and is acceptable.

Duke Position: Resolved per the Staff Position.

16. No separate effects comparison have been presented for the algebraic slip option and it would be prudent to request comparisons with the FRIGG tests before the approval of the algebraic slip option.

Staff Position: The algebraic slip option has been modified to include the Chexal-Lellouche drift flux model. Use of the Chexal-Lellouche drift flux model for BWR and PWR applications within the range of conditions covered by the steam-water database used to develop and validate the model is approved. The model has been qualified with data from a number of steady-state and two-component tests. While the small dimensions of the fuel assembly are covered, as noted previously in this safety evaluation, the data for large pipe diameters, such as reactor coolant system pipes, are not extensive and use of the Chexal-Lellouche model will need justification. Assessment work indicates that the model tends to under-predict the void profile in the range of 12 to 17 MPa. In addition, the accuracy of the model in the range of 7.5 to 10 Mpa, which covers BWR ATWS conditions, has not been fully demonstrated. Results of analyses using the model in these ranges must be carefully reviewed.

The Chexal-Lellouche correlation cannot be used in situations where CCFL is important unless validation for appropriate geometry and expected flow conditions is provided.

Duke Position:

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17. While FRIGG test comparisons have been presented for the dynamic slip option the issues concerning the Schrock-Grossman round tube data comparisons should be resolved before the dynamic slip option is approved. Plant comparisons using the option should also be required.

Staff Position: Assessment analyses (Reference 4), have shown that "the issues concerning the Schrock-Grossman round tube data comparisons" (actually the Bennett round tube data) are due to early prediction of CHF, which is nearly independent of the slip model used. Since the issue raised in the limitation is not related to the dynamic slip model, the limitation is considered to be resolved. The dynamic slip model is approved for use as given in condition (10).

Duke Position: Resolved per the Staff Position.

18. The nonequilibrium pressurizer model has no fluid boundary heat losses, cannot treat thermal stratification in the liquid region and assumes instantaneous spray effectiveness and a constant rainout velocity. A constant UA is used and flow detail within the component cannot be simulated. There will be a numerical drift in energy due to the inconsistency between the two-region and the mixture energy equations but it should be small. No comparisons were presented involving a full or empty pressurizer. Specific application of this model should justify the lack of fluid boundary heat transfer on a conservative basis.

Staff Position: The concern raised in this limitation of use is partially resolved in RETRAN-3D. Wall heat transfer can be included in the RETRAN-3D pressurizer model. Including wall heat transfer resolves this concern.

While the model does not directly account for thermal stratification, its effects can be included by use of normal nodes below the pressurizer volume. The user will have to justify the lack of thermal stratification or the use of normal nodes below the pressurizer should there be an indication that it would be important in the analysis.

The mixture and two-region energy equations are consistent for the implicit solution method where the mixture energy equation is used with the vapor-region energy equation. This eliminates inconsistency between the two-region and mixture energy equations and the concern regarding a potential drift in the region energies.

The staff notes that when a pressurizer fills or drains, a single region exists for which the normal pressure equation of state is used. Lack of numerical discontinuities in validation analyses of filling and draining pressurizers indicates that the model is functioning properly. It is the responsibility of the code user to justify any numerical discontinuity in the pressurizer during a filling or draining event.

The pressurizer model has options that require user-supplied parameters. Users must provide justification for these model parameters.

Duke Position: Duke is not proposing any changes in modeling the pressurizer with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA. The good modeling practices in the Staff Position are noted.

19. The non-mechanistic separator model assumes quasi-statics (time constant approximately few tenths of seconds) and uses GE BWR6 carryover/carryunder curves for default values. Use of default curves has to be justified for specific applications. As with the pressurizer a constant L/A is used. The treatment in the off normal flow quadrant is limited and those quadrants should be avoided. Attenuation of pressure waves at low flow/low quality conditions are not simulated well. Specific applications to BWR pressurization transients under those conditions should be justified.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not use the separator model.

20. The centrifugal pump head is divided equally between the two junctions of the pump volume. Bingham pump and Westinghouse pump data are used for the default single-phase homologous curves. The SEMISCALE MOD-1 pump and Westinghouse Canada data are for the degradation multiplier approach in the two-phase regime. Use of the default curves has to be justified for specific applications. Pump simulation should be restricted to single-phase conditions.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: As noted in the Staff Position, the basic model in RETRAN-3D is unchanged from RETRAN-02. Duke is not proposing any changes in modeling the reactor coolant pumps with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA, Section 2.2.6.2.

21. The jet pump model should be restricted to the forward flow quadrant as the treatment in the other quadrants is conceptually not well founded. Specific modellng of the pump in terms of volumes and junctions is at the user's discretion and should therefore be reviewed with the specific application.

Staff Position: Subsequent revisions of RETRAN-02 addressed this limitation. Since RETRAN-3D has the same model as RETRAN-02 MODOO3, and subsequent versions, their acceptance applies to RETRAN-3D.

Duke Position: Duke does not model BWR jet pumps.

22. The non-mechanistic turbine model assumes symmetrical reaction staging, maximum stage efficiency at design conditions, a constant UA and a pressure behavior dictated by a constant loss coefficient It should only be used for quasi-static conditions and in the normal operating quadrant.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not use the turbine model.

23. The subcooled void model is a nonmechanistic profile fit using a modification of EPRI recommendations for the bubble departure point It is used only for the void reactivity computation and has no direct effect on the thermal-hydraulics. Comparisons have only been presented for BWR situations. The model should be restricted to the conditions of the qualification data base. Sensitivity studies should be requested for specific applications. The profile blending algorithm used will be reviewed when submitted as part of the new manual (MODOO3) modifications.

Staff Position: The profile blending algorithm approved for RETRAN-02 MODOO3 is used in RETRAN-3D therefore this condition has been satisfied.

Duke Position: Resolved per the Staff Position.

24. The bubble rise model assumes a linear void profile, a constant rise velocity (but adjustable through the control system), a constant UA, thermodynamic equilibrium, and makes no attempt to mitigate layering effects. The bubble mass equation assumes zero junction slip which is contrary to the dynamic and algebraic slip model. The model has limited application and each application must be separately justified.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies. However, the layering effects encountered in RETRAN-02 can be eliminated using the RETRAN-3D stack model. This partially resolves the concern by resolving the layering limitation through use of the stack model.

Duke Position: As noted in the Staff Position, the basic model in RETRAN-3D is unchanged from RETRAN-02. Duke is not proposing any changes in using the bubble rise model with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA, Section 2.2.6.4. Duke does not currently stack bubble rise volumes, but if future modeling does, the stack model will be used.

25. The transport delay model should be restricted to situations with a dominant flow direction.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies. The appropriate application of the model is for one-dimensional flow. The user will have to justify use of this option in the absence of a dominant flow direction.

Duke Position: As noted in the Staff Position, the basic model in RETRAN-3D is unchanged from RETRAN-02. Duke is not proposing any changes in using the transport delay model with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA, Section 2.2.7.4. The limitation with applying this model without a dominant flow direction is well known and is avoided.

26. The stand-alone auxiliary DNBR model is very approximate and is limited to solving a one-dimensional steady-state simplified HEM energy equation. It should be restricted to indicating trends.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not use the auxiliary DNBR model.

27. Phase separation and heat addition cannot be treated simultaneously in the enthalpy transport model For heat addition with multidirectional, multifunction volumes the enthalpy transport model should not be used without further justification. Approval of this model will require submittal of the new manual (MODOO3) modifications.

Staff Position: A number of the simplifying assumptions in the RETRAN-02 enthalpy transport model have been eliminated in RETRAN-3D which now allows multiple inlet and outlet flows

and eliminates the simplifying assumptions related to mass distribution and pressure change effects. This condition has been adequately addressed.

Duke Position: Resolved per the Staff Position

28. The local conditions heat transfer model assumes saturated fluid conditions, one-dimensional heat conduction and a linear void profile. If the heat transfer is from a local condition volume to another fluid volume, that fluid volume should be restricted to a non-separated volume. There is no qualification work for this model and its use will therefore require further justification.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position:

29. The initializer does not absolutely eliminate all ill-posed data and could have differences with the algorithm used for transient calculations. A null transient computation is recommended. A heat transfer surface area adjustment is made and biases are added to feedwater inlet enthalpies in order to justify steady-state heat balances. These adjustments should be reviewed on a specific application basis.

Staff Position: The over-specified condition is identified by the RETRAN-3D steady-state input checking, resolving the concern regarding ill-posed data. The user must still run null transients to ensure that unwanted control or trip actions are not affecting the transient solution.

RETRAN-3D has available a low power steady-state steam generator initialization option that eliminates the heat conductor area change used in the RETRAN-02 initialization scheme. When this option is used, no adjustments are made to the heat transfer area and this specific concern is resolved. However, either the pressure or temperature is adjusted on the secondary side. These adjustments should be reviewed by the user on a specific application basis. The low power steady-state initialization option is approved for use.

Duke Position: Resolved per the Staff Position.

30. Justification of the extrapolation of FRIGG data or other data to secondary-side conditions for PWRs should be provided. Transient analysis of the secondary side must be substantiated. For any transients in which two-phase flow is encountered in the primary all the two-phase flow models must be justified.

Staff Position: The Chexal-Lellouche correlation is approved for use with PWR applications as stated in conditions (10) and (16). The user must justify choosing any other two-phase flow correlation.

Duke Position:

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31. The pressurizer model requires model qualification work for the situations where the pressurizer either goes solid or completely empties.

Staff Position: The pressurizer model is approved for use with filling and draining events as given in condition (18).

Duke Position: Resolved per the Staff Position

32. Transients which involve three-dimensional space-time effects such as rod ejection transients would have to be justified on a conservative basis.

Staff Position: The 3-dimensional kinetics model, as noted in limitation 1 above, satisfies this limitation.

Duke Position: Resolved per the Staff Position. Duke is not using the three-dimensional model.

33. Transients from subcritical, such as those associated with reactivity anomalies should not be run.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke is not running any transients from subcritical.

34. Transients where boron injection is important, such as steamline break will require separate justification for the user-specified boron transport model.

Staff Position: The generalized transport model was added to RETRAN-3D to provide the capability to track materials such as boron. Specific application of the model to steam line break transients must be justified by the user. The model is approved for use as given in condition (3).

Duke Position: The generalized transport model is unchanged from RETRAN-02 relative to its use for modeling boron transport. The generalized transport model was approved in the SER for RETRAN-02 MOD005.0 dated November 1, 1991. Duke's use of this model for Oconee

emergency boron injection is documented in topical report DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology," Revision 1, Section 15.3.1.1.3. This topical report was approved by the NRC with SERs dated October 1, 1998 and May 25, 1999. No further NRC review is necessary.

35. For transients where mixing and cross flow are important, the use of various cross flow loss coefficients has to be justified on a conservative basis.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke models mixing and cross flow in the reactor vessel during transients and accidents in which loop asymmetry is important. Duke's modeling for Oconee is documented in topical report DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology," Revision 1, Section 15.2.1.1. This topical report was approved by the NRC with SERs dated October 1, 1998 and May 25, 1999. No further NRC review is necessary.

36. ATWS events will require additional submittals.

Staff Position: RETRAN-3D is approved for PWR ATWS analyses as given in condition (13).

Duke Position: Resolved per the Staff Position.

37. For PWR transients where the pressurizer goes solid or completely drains, the pressurizer behavior will require comparison against real plant or appropriate experimental data.

Staff Position: The pressurizer model is approved for use with filling and draining events as noted in the discussion of conditions (18) and (31).

Duke Position: Resolved per the Staff Position

38. PWR transients, such as steam generator tube rupture, should not be analyzed for two-phase conditions beyond the point where significant voiding occurs on the primary side.

Staff Position: The use of slip models for PWR applications is approved for use as given in conditions (16) and (30).

Duke Position: In the Oconee steam generator tube rupture UFSAR Chapter 15 analysis significant voiding does not occur on the primary side. However, significant voiding can occur on the primary side for steam line break events. Duke's UFSAR Chapter 15 steam line break modeling for Oconee is documented in topical report DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology," Revision 1, Chapter 15. This topical report was approved by the NRC with SERs dated October 1, 1998 and May 25, 1999. Duke's UFSAR Chapter 6 steam line break mass and energy release modeling for Oconee is documented in topical report DPC-NE-3003-PA, "Mass and Energy Release and Containment Response Methodology," Chapter 5. This topical report was approved by the NRC with SER dated March 15, 1995. Modeling of two-

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] No further NRC review is necessary.

39. BWR transients were asymmetry leads to reverse jet pump flow such as the one recirculation pump trip, should be avoided.

Staff Position: As noted in the discussion of condition (21), this is resolved.

Duke Position: Duke does not model BWRs.

40. Organizations with NRC-approved RETRAN-02 methodologies can use the RETRAN-3D code in the RETRAN-02 mode without additional NRC approval, provided that none of the new RETRAN-3D models listed in the definition are used. Organizations with NRC-approved RETRAN-02 methodologies must obtain NRC approval prior to applying any of the new RETRAN-3D models listed above for UFSAR Chapter 15 licensing basis applications. Organizations without NRC-approved RETRAN-02 methodologies must obtain NRC approval for such methodologies or a specific application before applying the RETRAN-02 code or the RETRAN-3D code for UFSAR Chapter 15 licensing basis applications. Generic Letter 83-11 provides additional guidance in this area. Licensees who specifically reference RETRAN-02 in their technical specifications will have to request a Technical Specification change to use RETRAN-3D.

Duke Position: The submittal of DPC-NE-3000-P, Revision 3, includes the use of the [

] The SER on p. 33 also states that use of the "new control blocks added to improve functionality" requires NRC review and approval. The new control blocks in RETRAN-3D are the following:

ABS - Absolute value
F2D - Two-dimensional interpolation
RAT - Rate
STF - Second-order transfer function

None of these new control block models have yet been incorporated into any of the Duke RETRAN models used for licensing basis applications. However, use of some of these new control block models in the future is likely to enhance and simplify applications. Since all of these new control blocks consist of well-founded arithmetic and mathematical formulas, similar to the control blocks included in RETRAN-02, it is not understood why NRC approval prior to their use is necessary. Duke requests NRC approval to use the new RETRAN-3D control blocks for future applications consistent with their formulation.

41. RETRAN may be used for BWR ATWS subject to the following restrictions: The licensee must validate the chosen void model over the range of pressure, channel inlet flow, and inlet subcooling encountered during the transient that are outside the range of conditions for which assessment is available. Furthermore, the licensee should validate the choice of steam separator model and evaluate its use relative to steam separator performance data relevant to the conditions

present during the ATWS simulation. The licensee must also evaluate the uncertainties in the modeling. See Condition (16) and the Staff Position for related information.

Duke Position: Duke does not model BWRs.

42. The RETRAN-3D five-equation, or nonequilibrium, model uses flow regime maps and flow pattern dependent heat transfer and interfacial area models to simulate the heat and mass transfer processes between phases. A licensee wishing to apply the five-equation model will have to justify its use outside areas of operation where assessment has been documented. This may include either separate effects or integral systems assessment that cover the range of conditions encountered by the application of interest. An assessment of the uncertainties must also be provided. The model is approved subject to these conditions.

Duke Position: Duke does not use the five-equation model for licensing basis applications.

43. Assessment performed in support of use of RETRAN-3D must also address consistency between the RETRAN-3D calculations and any auxiliary calculations that are a part of the overall methodology, such as, departure from nucleate boiling or critical power ratio.

Duke Position: Duke uses results from RETRAN-3D analyses for input to other codes to perform core power distribution analyses, detailed core thermal-hydraulic analysis of the departure from nucleate boiling phenomenon, fuel rod and pellet thermal and mechanical behavior analyses, and containment thermal and structural response to high-energy line breaks. The details of these other methodologies have been submitted and approved by the NRC as appropriate. Any revisions to these methodologies, including any changes due to the use of RETRAN-3D in place of RETRAN-02, will be submitted for NRC review prior to their use for licensing basis applications.

44. The staff concludes that the lack of a detailed RETRAN-3D specific user guideline document mandates a statement on the user's experience and qualification with the code when analyses are submitted in support of licensing actions. This statement is expected to be consistent with the guidance of Generic Letter 83-11.

Duke Position: It is noted that Volume 3 of the EPRI RETRAN-3D documentation set has been enhanced subsequent to the NRC SER to include a significant amount of user guidelines regarding modeling option selection, in particular for the new RETRAN-3D models and options. Revision 3 to DPC-NE-3000-P fully describes Duke's use of the RETRAN-3D code for simulating the Oconee Nuclear Station with replacement steam generators. This revision is submitted for NRC review with the intent of maintaining the documentation of the Duke RETRAN methodology current, along with the main purpose of obtaining NRC review and approval for the transition from RETRAN-02 to RETRAN-3D for Oconee. This topical report revision extends Duke's response to Generic Letter 83-11. Duke's current level of RETRAN user experience is 15 engineers with a total of 144 years of experience with RETRAN-02 and RETRAN-03/-3D.

45. Assessment of the RETRAN-3D code for the models not explicitly approved in this safety evaluation will be the responsibility of the licensee or applicant. In addition, application of the

RETRAN-02 or RETRAN-3D codes for best estimate analysis of UFSAR Chapter 15 licensing basis events may require additional code and model assessment, and an evaluation of uncertainties to assure accurate prediction of best estimate response. This condition is based on the absence, in the best estimate analysis approach, of the conservative assumptions in traditional UFSAR Chapter 15 licensing basis analyses. For each use of RETRAN-3D in a licensing calculation, it will be necessary for a valid approach to assessment to be submitted, which is expected to include a PIRT for each use of the code and the appropriate assessment cases and their results. The scope of the PIRT and validation/assessment will be commensurate with the complexity of the application.

Duke Position: Duke has previously received NRC review and approval for application of the RETRAN-02 code to the licensing basis applications for non-LOCA transients and accidents for the Oconee Nuclear Station. The three RETRAN-related topical reports and associated NRC SERs supporting Oconee are:

DPC-NE-3000-PA, Revision 2, "Thermal-Hydraulic Transient Analysis Methodology", December 2000. SERs are dated 11/15/91 (Revision 0), 8/8/94 (Revision 1), and 12/27/95 (Revision 2)

DPC-NE-3003-PA, "Mass and Energy Release and Containment Response Methodology", November 1997. SER is dated 3/15/95

DPC-NE-3005-PA, Revision 1, "UFSAR Chapter 15 Transient Analysis Methodology, August 1999. SERs are dated 10/1/98 (Revision 0) and 5/25/99 (Revision 1)

These topical reports have all being revised and submitted for NRC review to address the Oconee replacement steam generators and use of the RETRAN-3D code for Oconee non-LOCA transient and accident analyses. Based on the close similarity of the replacement and original steam generators, the transient thermal-hydraulic behavior will be very similar. The only significant difference will be for the main steam line break analysis, in which the flow restricting orifices in the replacement steam generators steam outlet nozzles will effectively reduce the maximum break size and the blowdown rate.

[ ]

activity was presented and reviewed by the NRC during the review of earlier revisions of DPC-NE-3000. [ ] has been incorporated into the Duke version of the RETRAN-3D code as described in Revision 3 to DPC-NE-3000-P.

In summary, Duke has previously obtained NRC review for RETRAN-02 modeling of Oconee with the original steam generators. Substantial validation and assessments comparisons were associated with the previous revisions to DPC-NE-3000. The designs of the original and replacement steam generators are very similar, and the transient performance will be very similar except for the response to large steam line break accidents. Revision 3 describes the use of RETRAN-3D for modeling Oconee with replacement steam generators. [ ] Duke is not proposing to use this model for best-estimate licensing applications. The traditional conservative approach will continue to be used for licensing applications. A PIRT is not being submitted due to the previous NRC approval of the Duke RETRAN methodology topical reports, the limited scope of changes in the methodologies, the similarity of the designs of the new and replacement Oconee steam generators, the use of only one new RETRAN-3D model, and the assessment that has been performed to justify use of the one new model.

References

C-1 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

C-2 RETRAN-3D MOD003.1DKE, SDQA-30218-NGO, Duke Power, October 30, 2001

Attachment 4

Mass and Energy Release and Containment Response Methodology

DPC-NE-3003 (Non-Proprietary)
Revision 1

June 2002

Nuclear Engineering Division
Nuclear Generation Department
Duke Power Company

DPC-NE-3003 Non-Proprietary, Revision 1

Description and Technical Justification

Revision 1 to DPC-NE-3003 consists of changes necessary to model the replacement steam generators with RETRAN-3D and RELAP5, a new approach to address water carryout during main steam line break, an addition of the GOTHIC 7.0 code for containment analysis, modeling enhancements, technical and editorial changes to modernize and update the methodology since Revision 0 was submitted in 1993, and an evaluation of compliance with the NRC's SER on RETRAN-3D. Each of the revisions is described in detail, and model revisions are supported by technical justification. The RETRAN-3D modeling has been reviewed by Computer Simulation & Analysis (CSA), Inc., the developer of the RETRAN-3D codes. There are currently no other organizations using RETRAN-3D for modeling B&W-designed reactors, so a peer review other than by the code vendor was not possible. The GOTHIC 7.0 content of the report has been reviewed by Numerical Applications, Inc., the developer of the FATHOMS and GOTHIC codes.

The Oconee replacement steam generators (ROTSGs) are being manufactured in Canada by Babcock & Wilcox Canada (BWC). The ROTSG are quite similar functionally to the original OTSGs (Figure 1), and are characterized as a like-for-like replacement. The design differences that are important for thermal-hydraulic modeling are 1) flow-restricting orifices in the steam outlet nozzles, 2) Inconel-690 tubes, 3) 15,631 vs 15,531 tubes, 4) thinner pressure vessel / wider downcomer, 5) thinner tubesheets resulting in 3.625 inch longer heated tube length, 6) 1.2% greater heat transfer area, and 7) more water in the steam generator. Of these design changes only the flow-restricting steam exit nozzles have a significant impact on the UFSAR transient and accident analyses. These nozzles reduce the effective size of steam line breaks during blowdown of a steam generator to 1.804 ft^2. This design feature significantly changes the steam line break mass and energy release. The other design changes will not cause a significant change in plant response during transients and accidents. For that reason the modeling of the ROTSGs is similar to the modeling of the OTSGs, with changes limited to those described herein.

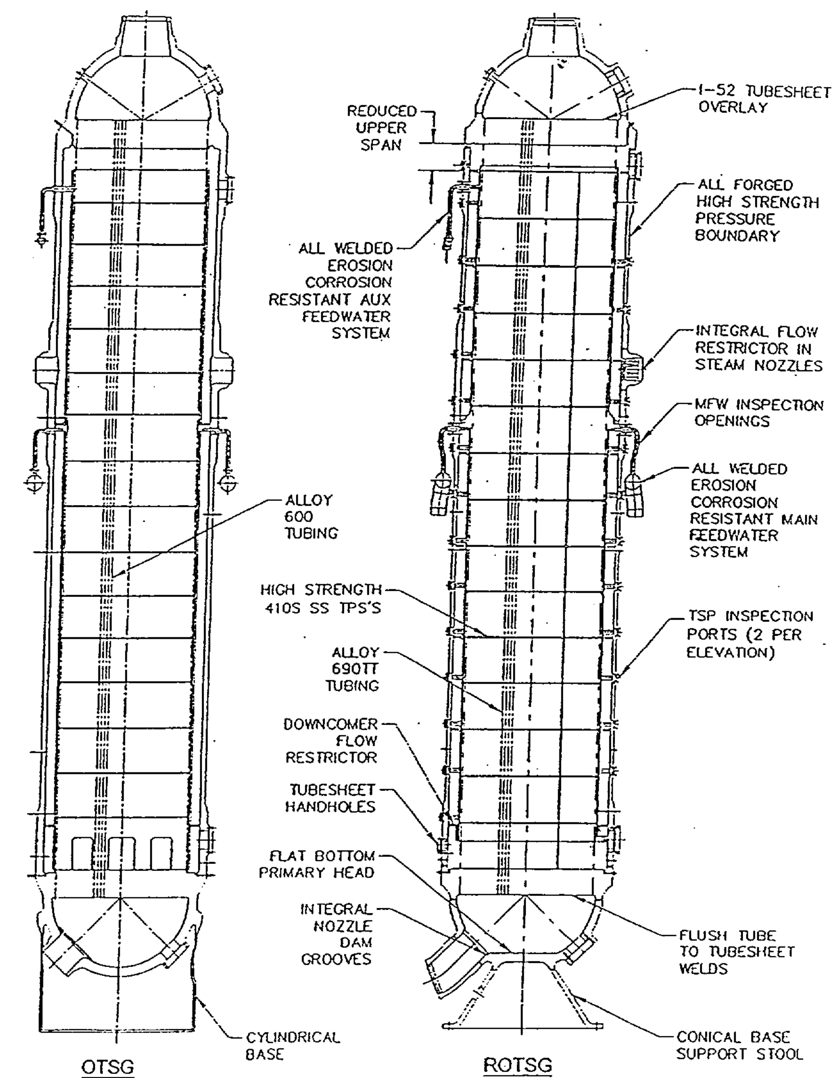
The original steam line break mass and energy release analysis methodology was based on the Electric Power Research Institute's RETRAN-02 code. The next generation in the RETRAN family of codes, RETRAN-3D, was approved by the NRC in the SER dated January 25, 2001. Revision 1 to DPC-NE-3003 describes the modeling for the Oconee steam line break mass and energy release analysis with RETRAN-3D replacing RETRAN-02. [

] In addition, a new modeling approach was developed by Duke and CSA to conservatively account for water carryout during steam line breaks. All of the details of the transition to using RETRAN-3D instead of RETRAN-02 for Oconee modeling are presented. The conditions and limitations in the NRC's SER for RETRAN-3D are also addressed.

The modeling enhancements and technical and editorial revisions that modernize and update the methodology are not necessarily associated with either the ROTSGs or the transition to

RETRAN-3D. These changes are technical enhancements or are necessary to maintain the content of the report consistent with the current plant design, as well as to correct errors. No technical justification is necessary for editorial revisions and corrections. The changes are presented in the order that they appear in DPC-NE-3003. The analysis inputs and results presented in Chapters 3-6 are a demonstration of the Revision 0 methodology. New Appendix A describes the Oconee ROTSG steam line break mass and energy release methodology. New Appendix B addresses the limitations and conditions in the RETRAN-3D SER as relates to the application to steam line break mass and energy release. New Appendix C details the use of GOTHIC 7.0 for the Oconee containment response analysis methodology.

Figure 1 OTSG / ROTSG Comparison



Changes and Technical Justification

Cover Page and Frontal Pages

1. The revision and date will be revised

2. The table of contents and the lists of table, figures , and acronyms will be updated

Chapter 1

3. Page 1-1, second paragraph, revise to indicate the correct location in the UFSAR of the analyses associated with this topical report (editorial)

   Change: "Chapter 15 of the"

   To: "Chapter 15 (now Section 6.2)"

4. Page 1-3, new text at bottom, add the following descriptions of the existing supplements and the new Revision 1 appendices (editorial)

   Supplement 1 was added with the November 1997 version of Revision 0 of the report to present updated steam line break mass and energy release and containment response analysis results that supercede the results in Chapters 5 and 6 of the original report.

   Supplement 2 was added with the November 1997 version of Revision 0 of the report to present validation analyses comparing the FATHOMS prediction to data from the CVTR facility.

   Appendix A was added in Revision 1 to describe the replacement steam generator (ROTSG) steam line break mass and energy release modeling with RETRAN-3D (Reference 1-10).

   Appendix B was added in Revision 1 to address the RETRAN-3D SER (Reference 1-11) conditions and limitations as relates to the replacement steam generator (ROTSG) steam line break mass and energy release modeling with RETRAN-3D.

   Appendix C was added in Revision 1 to include the use of the GOTHIC 7.0 code (Reference 1-12) for containment response analysis.

5. Section 1.1: New references added and existing references updated to current revisions. (editorial)

   1-5 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002

   1-10 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI, NP-7450(A), Volumes 1-4, Revision 5, July 2001

1-11 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

1-12 GOTHIC Containment Analysis Package Version 7.0, NAI-8907-02, Revision 13, July 2001

Chapter 2

6. Section 2.1.2, p. 2-6, Simulation Model, is revised to include a new subsection "Main Feedwater Piping" to describe the inclusion of additional nodes in the RELAP5 model to include main feedwater piping. (model revision)

   Insert the following new subsection at the bottom:

   "Main Feedwater Piping

   The RELAP5 nodalization is revised to include the main feedwater piping between the last check valve and the steam generator. This enables modeling flashing of the main feedwater in the piping if the steam generator pressure decreases low enough for the flashing to occur. Should this occur additional hot water will be expelled into the steam generator with the potential to increase secondary-to-primary heat transfer. This additional modeling detail is necessary to accurately model the main feedwater boundary condition."

   Technical Justification: Steam generator depressurization to below the saturation pressure of the hot water in the main feedwater piping will cause some of the water to flash to steam and expel water into the steam generators. For LOCA mass and energy releases it is possible that the steam generators will depressurize sufficiently for this flashing phenomenon to occur. Therefore it is necessary for the RELAP5 model to include the main feedwater piping that may flash. The results of the analyses have shown this to be a very minor effect, but it is modeled for completeness.

7. Section 2.1.2, p. 2-6, Simulation Model, is revised to include a new subsection "ROTSG Model" to describe the changes in the RELAP5 model to represent the replacement steam generators. (model revision)

   Insert the following new subsection at the bottom:

   "ROTSG Model

   The replacement steam generators (ROTSGs) are modeled in RELAP5 with the same nodalization as the original steam generators. All of the RELAP5 model input data have been recalculated for the ROTSGs based on data supplied by B&W Canada, the manufacturer. Refer to Appendix A for a description of the design differences."

   Technical Justification: The Oconee replacement steam generators are very similar to the original steam generators. The main differences are dimensional or material, such as the steam outlet nozzles being equipped with flow-restricting orifices, Inconel-690 tube alloy, slightly longer tubes, thinner pressure vessel, etc. Based on these limited design changes the existing RELAP5 nodalization was judged to be suitable for the ROTSGs also. All of the input data were recalculated based on data supplied by B&W Canada (BWC), the manufacturer. The initial conditions provided by BWC were used to determine the conservative initial conditions for the applications in this report. This model building activity and the integration with the original RELAP5 model have achieved the desired results.

8. Page 2-22 was inadvertently replaced by Page 3-22 in the 1997 published version. The corrected page 2-22 is the next page in this attachment (editorial)

9. New Section 2.5 "RETRAN-3D Code Description". (model revision)

"2.5 RETRAN-3D Code Description

RETRAN-3D (Reference 2-23) was developed by Computer Simulation & Analysis, Inc. for EPRI to enhance and extend the simulation capabilities of the RETRAN-02 code. RETRAN-3D has many new and enhanced capabilities relative to RETRAN-02, in particular, a 3-D kinetics core model, improved two-phase models, an improved heat transfer correlation package, and an implicit numerical solution method. Most of the capabilities of the RETRAN-02 code have been retained within RETRAN-3D as options, except for a limited number of models and correlations that were not in use. The NRC SER for RETRAN-3D is dated January 25, 2001 (Reference 2-24). The SER includes limitations and conditions on the use of the code for licensing applications. Refer to Appendix A for details on the application of RETRAN-3D for steam line break mass and energy release analyses. Appendix B is an evaluation of the conditions and limitations in the RETRAN-3D SER."

10. New Section 2.6 "GOTHIC 7.0 Code Description". (model revision)

2.6 GOTHIC 7.0 Code Description

The GOTHIC 7.0 code (Reference 2-25) was developed by Numerical Applications, Inc. for EPRI as the next version of the GOTHIC family of codes for thermal-hydraulic analysis of nuclear power plant containment and auxiliary buildings. The GOTHIC code was developed from the FATHOMS code (Reference 2-19), which in turn was developed from the COBRA-NC code (Reference 2-20). Similar to FATHOMS, GOTHIC 7.0 solves the conservation equations for mass, energy, and momentum for multi-component two-phase flow. Refer to Appendix C for details on the application of GOTHIC 7.0 for LOCA and steam line break containment response analyses.

11. Section 2.4 Referenced renumbered to 2.7 and new references and revisions to existing references added (editorial)

2-18 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002

2-23 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI, NP-7450(A), Volumes 1-4, Revision 5, July 2001

Refer to Change #8

2-24 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

2-25 GOTHIC Containment Analysis Package Version 7.0, NAI-8907-02, Revision 13, July 2001

Chapter 3

12. Section 3.1, p. 3-2, second paragraph, change the termination of the RELAP5 analysis from 30 minutes to the end of the BWST injection phase (model revision)

    Change: "quasi-steady-state conditions have evolved in the RCS ($\approx$30 minutes)."

    To: "the end of the cold leg injection phase."

    Technical Justification: With further experience in applying the methodology it has become apparent that the appropriate time for terminating the RELAP5 phase of the analysis is at the transition from the BWST injection phase to the sump recirculation phase, not necessarily at 30 minutes. At that point in the LOCA scenario the system has evolved into a nearly steady-state boiling pot mode of core cooling, and the transition to the use of the BFLOW code for the continuation of the mass and energy release calculation can be made. The transition to the sump recirculation phase also involves a step change in the ECCS suction temperature. Making the code transition at this point in time is consistent with a transition in the cooling mode.

13. Section 3.1, p. 3-2, second paragraph, clarify that the BFLOW code is only used for cold leg break LOCA analysis, and to describe what is used for hot leg break LOCAs. (model revision)

    Change: "The BFLOW code is used to calculate the mass and energy release for the remainder of the analysis."

    To: "The BFLOW code is used to calculate the mass and energy release for the remainder of the analysis for cold leg break LOCAs. RELAP5 is run for hot leg LOCAs until the break flow has reached a quasi-steady condition. The remainder of the mass and energy release is performed in FATHOMS using an input boundary condition based on decay heat."

    Technical Justification: The BFLOW code is designed to calculate the long-term mass and energy release for cold leg break LOCAs. This was not mentioned in the original report since the focus of the report was modeling cold leg LOCAs. This change is made for clarification purposes. The hot leg break long-term mass and energy release is calculated in the FATHOMS code. This method consists of adding decay heat through boundary condition data to heat the ECCS water originating in the sump after it has been cooled by the LPI heat exchangers. This simple approach is appropriate because hot leg break LOCAs evolve into a simple process by which water is pumped into the reactor vessel where it flows up through the core and absorbs decay and sensible heat before flowing out the break.

14. Section 3.2, p. 3-4, Pressurizer Level, the basis for the initial pressurizer level has been updated for consistency with technical specifications. (model revision)

Change: "The level control system normally maintains pressurizer level at the 220 inch setpoint. However, the only operational restriction imposed on pressurizer level during power operation is to trip the reactor manually if level reaches 375 inches. High level and high-high level alarms exists at 260 and 315 inches to warn the operator of a high level condition. Although an allowance of 25 inches has been determined to bound the level uncertainty, an initial level of 245 inches (220 + 25) is less than the setpoint value for both high level alarms. The initial pressurizer level is assumed to be 315 inches since it is reasonable to believe that the operators will respond to the high level alarms and act to maintain pressurizer level below this value."

To: "The level control system normally maintains pressurizer level at the 220 inch setpoint. High level and high-high level alarms exists at 260 and 315 inches to warn the operator of a high level condition. Technical specifications require pressurizer level to be <285 inches. The uncertainty on the level instrument is 25 inches. Therefore, an initial condition of 310 inches, corresponding the technical specification value plus uncertainty is assumed in the analyses."

Technical Justification: The technical specifications include limits on the ranges of parameters for operation to ensure that the licensing basis safety analyses remain bounding. Assuming the maximum allowed technical specification value for initial pressurizer level plus instrument loop uncertainty is appropriate for this application.

15. Section 3.2, p. 3-5, Steam Generator Operating Level, is changed to indicate that the current mass values are only applicable to the original steam generators, and that a conservative high mass is assumed for the replacement steam generators. (model revision)

Change: "The nominal full power steady-state mass is [ ] per steam generator."

To: "The nominal full power steady-state mass is [ ] per steam generator for the original steam generators."

Change: "An initial steam generator mass of approximately [ ]. . . "

To: "An initial steam generator mass of approximately [ ] for the original steam generators . . . "

Insert at the bottom: "A conservative high initial mass is assumed for the replacement steam generators."

Technical Justification: The methodology for both the original and the replacement steam generators assumes a high initial mass. This maximizes the energy transferred from the secondary to the primary during the LOCA mass and energy release, and is a conservative assumption. This revision clarifies that the mass numbers in the report are applicable to the original steam generators only.

16. Section 3.2, p. 3-6, Safety Injection Tanks, is revised to "Core Flood Tanks", and to change the water temperature from 120°F to 130°F to be consistent with the bounding tank water temperature. (editorial)

    Change: "Safety Injection Tanks"

    To: "Core Flood Tanks"

    Change: "120°F"

    To: "130°F"

    Technical Justification: The ECCS accumulators at Oconee are referred to as the core flood tanks, and so the title of this section is revised. Although the temperature of the core flood tank water is considered an input value, since the 120°F value was stated in the topical report it is updated to 130°F, the value currently assumed to bound plant operating conditions.

17. Section 3.2, p. 3-7, RCS Flow, is changed to indicate that the current flow values are only applicable to the original steam generators, and that high and low RCS flow are analyzed as a sensitivity parameter for the replacement steam generators. (model revision)

    Insert new paragraph at the bottom:

    "The above discussion is applicable to the methodology used in the analysis of the original steam generators. For the replacement steam generator analyses RCS flow is a sensitivity parameter and both bounding high and low initial RCS flow values are analyzed to determine the limiting case."

    Technical Justification: The methodology for the original steam generators assumed low initial flow was conservative based on the justification presented. The flow numbers are associated with the original analysis. The application of the methodology for the replacement steam generators has been revised to consider RCS flow as a sensitivity parameter, and analysis are performed with both bounding high and bounding low RCS flow values to determine the limiting case. Although RCS flow is not an extremely important initial condition, this approach ensures that the limiting case is analyzed.

18. Section 3.3.1.1, p. 3-8, RCS and Steam Generator Level, the pressurizer level input value is changed from 315 inches to 310 inches. (editorial)

    Change: "315 inches"

    To: "310 inches"

    Technical Justification: See Section 3.2 item above for details.

19. Section 3.3.1.1, p. 3-9, Core Stored Energy, the fuel temperature values are those assumed in the original analysis. New analyses use bounding values for the current fuel design. (editorial)

    Insert new paragraph at the bottom:

"The above fuel temperatures were assumed in the original analyses. New analyses use bounding values for the current fuel design."

Technical Justification: Fuel temperature is considered an input value in the methodology. The values presented in the report were bounding values for the fuel designs used at that time. The methodology requires bounding high fuel temperatures for conservatism. New analyses use bounding high values for the fuel designs currently in use.

20. Section 3.3.1.1, p. 3-10, Main Feedwater Line requires revision to update the modeling when the flow is realigned to the upper auxiliary header following trip of the reactor coolant pumps. Also, the subsection title "Main Feedwater Line" is revised to "Main Feedwater System". (model revision)

    Change: "Main Feedwater Line" to "Main Feedwater System"

    Insert the following at the bottom: "The Main Feedwater System is realigned to feed through the upper auxiliary feedwater header following the trip of all reactor coolant pumps. Flow through this alignment is restricted to much lower flowrates to prevent exceeding the steam generator tube flow-induced vibration limit."

    Technical Justification: The Main Feedwater System is realigned by the Integrated Control System to feed through the upper auxiliary header whenever all four reactor coolant pumps are tripped. This realignment is intended to promote the natural circulation mode of heat transfer. Flow through the upper header is restricted to not exceed the steam generator tube flow-induced vibration limit. This results in less feedwater flow following the realignment. This modeling approach is consistent with the plant design.

21. Section 3.3.1.2, p. 3-12, Borated Water Storage Tank (BWST) Level and Temperature, the last sentence in the first paragraph is deleted since it is no longer correct. (editorial)

    Delete: "With a maximum BWST level uncertainty of 20.2 inches, a minimum initial level of 44.3 feet is assumed."

    Technical Justification: The technical specifications surveillance on BWST level has been revised so that the 46 feet level requirement must be exceeded by the instrument uncertainty. Therefore, the minimum actual level is 46 feet, and that is what is now assumed in the analysis.

22. Section 3.3.1.2, p. 3-13, Steam Generator Level, the 10.5% value for level instrument error is deleted. (editorial)

    Delete: "(10.5%)"

    Technical Justification: The 10.5% steam generator level instrument error is an input to the analysis. Specifying a value is not required in the topical report. A bounding instrument error value is assumed in the analyses.

23. Section 3.3.2.1, p. 3-15, Fission Heat, clarify that ECCS boron concentrations are actually in the COLR, and delete a sentence that has extraneous information on BWST boron concentrations. (editorial)

    Change: "The minimum ECCS boron concentration required by Technical Specifications ensures . . . "

    To: "The minimum ECCS boron concentration required by technical specifications and specified in the COLR ensures . . . "

    Delete: A minimum BWST boron concentration of 1950 ppm is required by the COLR as referenced by Technical Specifications (Note: Higher boron concentrations than 1950 ppm exist in current COLRs).

    Technical Justification: These changes are editorial in nature and simply clean up the text by deleting extraneous content and making clarifications.

24. Section 3.3.2.1, p. 3-18, first paragraph, Metal-Water Reaction Rate, the methodology has been changed to assumed a whole core oxidation rate of 1.0% rather than an analysis-specific result. (model revision)

    Change: "0.557% which is presented in Reference 3-6, Section 8. This fraction corresponds to a total energy release of 3.134 x 10^6 Btu."

    To: "1.0%, consistent with the regulatory limit."

    Technical Justification: The original text states a metal-water reaction rate of 0.557% based on the then-current LOCA analysis of record. The methodology has been revised to assume a value of 1.0%, which is the regulatory limit. This simplifies the methodology and avoids the possibility of an invalid input to the methodology in the future. The sentence also give the reference for the 0.557% value and a conversion to Btu units. Neither of these are needed with the new approach and they are deleted.

25. Section 3.3.2.2, p. 3-20, ECCS Injection, second paragraph, the modeling of LPI flow is revised to be consistent with current design and operating procedures (plant design change)

    Change: " Injection flow will be available from two HPI pumps and one LPI pump following the failure of a 4160V switchgear. The injected HPI flow is represented as a function of RCS back pressure. LPI flow is assumed to be constant at 3000 gpm based upon throttling guidance in the Emergency Operating Procedure. This value is biased by +311 gpm to account for instrument uncertainties. Instrument error is added to the assumed LPI flow since the greater flow will increase the core reflood rate and release energy to the containment sooner than a lesser flow would."

    To: "Injection flow will be available from two HPI pumps and one LPI pump following the failure of a 4160V switchgear. The injected HPI flow is represented as a function of RCS back pressure. LPI flow is assumed to be a function of RCS backpressure until operator action is assumed based upon throttling guidance in the emergency operating procedure. The throttled flow value is biased to account for instrument uncertainty."

Technical Justification: Throttling of LPI flow as directed by the Oconee Emergency Operating Procedure has changed since the original methodology was submitted. The above change reflects the current procedure guidance. These changes in the methodology are necessary to maintain consistency between the plant design and procedures and the UFSAR analyses.

26. Section 3.3.2.2, p. 3-21, Reactor Coolant Pumps, the two-phase head degradation model has been changed from the RELAP5 Semiscale pump model, [ ] model. (model revision)

    Change: " The RELAP5 two-phase pump head degradation model is used to model RCP performance in a saturated system. Two-phase head degradation characteristics are assumed to be similar to the Semiscale pump."

    To:

    [ ]

    Technical Justification:

    [ ]

27. Section 3.3.3.1, p. 3-22, Fission Heat, the fission power history from the B&W CRAFT2 LOCA analysis is no longer used. Also, the Metal-Water Reaction modeling is now the same as in the short-term analysis methodology. The methodology of Section 3.3.2.1 is used instead. (model revision)

    Delete: All text in Section 3.3.3.1.

    Insert: "Refer to Section 3.3.2.1."

    Technical Justification: The original long-term LOCA mass and energy release methodology used fission heat results from a B&W CRAFT2 LOCA simulation for Oconee as an input rather than actually calculating the post-LOCA core power response due to changes in reactivity during the event. In Revision 1 of DPC-NE-3003 the CRAFT2 results are no longer used. The RELAP5 modeling described in Section 3.3.2.1 is used instead, since it will reflect the initial and boundary conditions, as well as the transient prediction of the RELAP5 code. Also, the metal-water reaction of 1.0% as described in Section 3.3.2.1 is also used. Rather than repeat the text, a reference to Section 3.3.2.1 suffices. This is an improvement in the methodology.

28. Section 3.3.3.2, p. 3-23, Assumptions, is revised to include discussion of hot leg break locations. (model revision)

    Change: " The long-term large break containment analysis considers only a single break size and location: a double-ended guillotine break located at the A1 cold leg pump discharge."

    To: "The long-term break containment analysis considers both cold leg pump discharge and hot leg breaks. "

    Technical Justification: Additional analysis experience since the submittal of the original topical report has shown a need to perform hot leg break analyses for other design purposes. The methodology is revised to include hot leg breaks in addition to cold leg breaks.

29. Section 3.3.3.2, p. 3-24, Break Location, is revised to include discussion of hot leg break locations. (model revision)

    Change: "As described previously, the break size and location chosen for this analysis is a double-ended guillotine break (8.55 ft^2) located at the A1 cold leg pump discharge. The basis for choosing a cold leg break as opposed to a hot leg break is obvious once the characteristics of each break are considered."

    To: "The break size and location is dependent upon the analysis result of interest. For the limiting containment EQ response, the limiting break location is a double-ended guillotine break (8.55 ft^2) at the reactor vessel outlet nozzle. For the maximum containment sump water temperature analysis, the limiting break location is a double-ended guillotine break (14.1 ft^2) located at the reactor vessel outlet nozzle."

    Change: "Naturally, when the break site is flooded, decay heat will be transferred to containment in the liquid phase resulting in a less severe containment response."

    To: "Naturally, when the break site is flooded, decay heat will be transferred to containment in the liquid phase resulting in a less severe containment pressure response, but a higher sump water temperature."

    Technical Justification: These revisions are necessary to expand the discussion to include hot leg breaks and explain associated LOCA phenomena.

30. Section 3.3.3.2, p. 3-24, BWST Depletion - ECCS and Containment System Flow Rates, is revised to add information on the sensitivity of the results to BWST depletion rates. (model revision)

    Change: " Assumptions which deplete the BWST and cause an early switch to sump recirculation are therefore conservative with respect to long term mass and energy releases. BFLOW analyses substantiate this presumption. The break steam flow rate calculated by BFLOW is much more sensitive to ECCS subcooling than it is ECCS flow rate. Consequently, depleting the BWST rapidly more than compensates for the slight reduction in break steaming rate which will occur with the higher ECCS flow rates assumed to deplete the BWST.

The objective of this boundary condition is to then deplete the BWST as quickly as possible to maximize the time that sump recirculation is aligned. This is accomplished by assuming conservatively high flow rates "

To: "BFLOW analyses substantiate that the break steam flow rate is much more sensitive to ECCS subcooling than it is ECCS flowrate. Consequently, depleting the BWST rapidly more than compensates for the slight reduction in break steaming rate which will occur with the higher ECCS flow rates assumed to deplete the BWST. However, higher ECCS flow rates tend to enhance LPI cooler performance. These considerations are included in the selection of the ECCS boundary conditions."

The objective of this boundary condition is to then deplete the BWST to determine the time that sump recirculation is aligned. This is accomplished by assuming conservatively high or low flow rates "

Technical Justification: These revisions are necessary to expand the text to include hot leg break locations and the maximum sump temperature analysis, which can involve different assumptions regarding the BWST depletion rate in order to ensure a conservative analysis.

31. Section 3.3.3.2, p. 3-26, BWST Depletion - ECCS and Containment System Flow Rates, is revised to include methodology changes to reflect the current station design and procedures. (model revision)

    Change: " The assumed single failure of a 4160V switchgear will eliminate all but one LPI and one RBS pump. The constant LPI flow assumed is identical to the peak pressure case: 3000 gpm +311 gpm to account for instrument errors. RBS flow is modeled in this analysis to calculate the BWST depletion rate. The Emergency Operating Procedure instructs the operator to limit RBS flow to ≤1500 gpm per train. When the post-accident uncertainty associated with the spray instrument is considered, it is possible for the spray pump to deliver 1643 gpm while the instrumentation reads 1500 gpm. The RBS System is assumed to deplete the BWST at a constant rate of 1643 gpm. It should be noted that an inconsistency is introduced here between the spray flow assumed in the BWST depletion calculation and the flow assumed in the FATHOMS analysis. Instrument errors are subtracted from 1500 gpm to yield the assumed spray flow in the FATHOMS analysis because this is more conservative with respect to the long-term containment response. This 286 gpm discrepancy is accounted for in the FATHOMS analysis by assuming that the fluid is spilled directly to the sump in order to conserve mass inventory in containment.

    The switch to sump recirculation will occur at a nominal BWST level of 6 feet. An uncertainty is 20.2 inches is associated with the BWST level indication. Although this error is applied in the negative sense when determining the initial BWST level (see Section 3.3.1.2, BWST Level and Temperature), it is applied in the positive sense when determining the setpoint to align sump recirculation. When the assumed setpoint of 7.68 feet is reached in the analysis, the switch to sump recirculation is assumed to occur instantaneously (i.e. no delay for valve stroke times and operator actions)."

To: "The assumed failure of a 4160V switchgear will eliminate all but one LPI and one RBS pump. LPI flow is assumed to be a function of RCS backpressure until operator action is assumed based upon throttling guidance in the Emergency Operating Procedure. The throttled flow value is biased to account for instrument uncertainties. RBS flow is modeled in this analysis to calculate the BWST depletion rate. The RBS System is assumed to deplete the BWST at a constant rate using a conservative flow rate. It should be noted that consistent assumptions are made between the spray flow assumed in the BWST depletion calculation and the flow assumed in the FATHOMS analysis. BWST water that is assumed to be spilled directly to the sump is accounted for in the FATHOMS analysis in order to conserve mass inventory in containment.

The emergency operating procedure instructs the operator to switch to sump recirculation at a nominal BWST level of 6 feet. An uncertainty adjustment associated with the BWST level indication is made to conservatively minimize the amount of available BWST inventory. Although this uncertainty adjustment is applied in the negative sense when determining the initial BWST level (see Section 3.3.1.2, BWST Level and Temperature), it is applied in the positive sense when determining the setpoint to align sump recirculation. When the assumed setpoint is reached in the analysis, the switch to sump recirculation is assumed to occur instantaneously (i.e. no delay for valve stroke times and operator actions)."

Technical Justification: Since the original methodology was submitted the station design and procedures have been revised. This has necessitated revising the methodology for the depletion of the BWST and the associated operator actions and setpoints. The revisions stated above integrate the original methodology that remains valid and the new aspects of the methodology that reflect the changes at the plant. In addition, the revised methodology is focused on maintaining consistency between the mass and energy release elements and the containment response elements of the methodology.

32. Section 3.3.3.4, p. 3-31, Overview, the modeling transition no longer occurs at 30 minutes but rather at the end of the RELAP5 analysis. (model revision)

    Change: "Because RELAP5 is not used to predict the mass and energy release beyond 30 minutes, a conservative method must be derived to release this stored energy during the FATHOMS analysis."

    To: "At the end of the RELAP5 analysis a conservative method must be derived to release this stored energy during the FATHOMS analysis."

    Technical Justification: With further experience in applying the methodology it has become apparent that the appropriate time for terminating the RELAP5 phase of the analysis is at the transition from the BWST injection phase to the sump recirculation phase, not necessarily at 30 minutes. At that point in the LOCA scenario the system has evolved into a nearly steady-state boiling pot mode of core cooling, and the transition to the use of the BFLOW code for the continuation of the mass and energy release calculation can be made. The transition to the sump recirculation phase also involves a step change in the ECCS suction temperature. Making the code transition at this point in time is consistent with a transition in the cooling mode.

33. Section 3.7: Two references deleted and one new reference added. (editorial)

Reference 3-8 - Deleted

Reference 3-9 - Deleted

Reference 3-10 BWNT LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants, BAW-10192-PA, Framatome Advance Nuclear Products, Revision 0, June 1998

Chapter 4

34. Section 4.1, p. 4-2, Analysis Overview, second paragraph, the containment backpressure is now modeled as a boundary condition rather than with a simple containment model within the RELAP5 model. (model revision)

Change: " A model of the Reactor Building is added to each small break analysis to provide a means for estimating these boundary conditions."

To: "Containment pressure and sump water temperature are input to RELAP5 as boundary conditions based on FATHOMS analysis results."

Technical Justification: For small break LOCA mass and energy release analyses the simulation can last many hours. Consequently the containment backpressure on the break and the containment sump water temperature are important boundary conditions. In particular the containment sump water temperature is much hotter than the BWST temperature, and has a direct impact on the mass and energy release. In the original methodology a simple containment model was develop within the RELAP5 model. This was only an approximate approach but was suitable for the intended purpose. In Revision 1 this approach is replaced by using boundary conditions (pressure and sump water temperature) based on results from FATHOMS analyses. The outputs from FATHOMS analyses for similar break sizes are used, with attention to ensuring applicability to the RELAP5 case.

35. Section 4.1, Long-Term Mass and Energy Release, first paragraph on p. 3, is revised to update the description of the modeling of the stored energy during the transition from RELAP5 to FATHOMS. (model revision)

Change: "The initial fluid conditions in each node is determined by averaging fluid conditions in the corresponding RELAP5 nodes. The RELAP5 heat structures are also collapsed into three heat slabs consistent with the simplified node boundaries. The average temperature in each heat slab is similarly determined by averaging the stored energy in all corresponding RELAP5 heat structures. "

To: "The initial fluid conditions in each node are determined by averaging fluid conditions in the corresponding RELAP5 nodes. The stored energy in the RELAP5 primary system heat structures are modeled by three FATHOMS heater components consistent with the simplified node boundaries. The stored energy in the RELAP5 secondary fluid and secondary heat structures are modeled by two additional FATHOMS heater components. The FATHOMS heater component models the stored energy in the

RELAP5 components as an average heat flux for a specified time period boundary condition."

Technical Justification: The transition from RELAP5 to FATHOMS for the small break LOCA mass and energy release modeling requires consideration of the stored energy at the time of the transition. This modeling has been revised as follows in Revision 1. The stored energy in the secondary system fluid and structural metal is now included in Revision 1. This is done for completeness in modeling all possible energy sources. All of the stored energy modeling now uses the FATHOMS heater component model. The various stored energy sources in RELAP5 are converted into average heat flux for a specified time period boundary conditions. This is an improvement and a simplification relative to the original methodology that modeled the stored energy as heat slabs and required an appropriate heat transfer coefficient to be modeled.

36. Section 4.3.1, Fission Heat, p. 4-3, is replaced in its entirety to use simpler modeling that has the same result. (model revision)

"Fission Heat

The RELAP5 point kinetics model is used to calculate delayed neutron power as a function of time. Upon reactor trip a negative reactivity insertion equal to the technical specification minimum shutdown margin, currently 1.0% $\Delta k/k$, using a normalized rod worth vs. position curve is modeled. The reactor is assumed to trip on the first RPS setpoint encountered. This setpoint is generally the variable low RCS pressure trip which is modeled with the equation given in the technical specifications. Instrument uncertainties are also applied to this equation. The rod insertion rate is based upon Technical Specification values. For conservatism, a delay is added so that the rods begin to fall 0.7 seconds after the trip signal is received. No thermal feedback is modeled. This approach is consistent with the core reload design process which ensures that the minimum shutdown margin is maintained post-trip for LOCA-type events. This is a conservative model since greater than the minimum shutdown margin always exists."

Technical Justification: Both the original and the Revision 1 methodology conservatively limit the negative reactivity insertion on reactor trip to the technical specification shutdown margin. In the original methodology thermal feedback was calculated, but then the total reactivity was limited to the minimum shutdown margin. In effect, modeling the thermal feedback was not actually contributing to the analysis result. In Revision 1 the thermal feedback has been deleted.

37. Section 4.3.1, Total Rod Worth, p. 4-4, is deleted in its entirety and replaced with text to indicate that the total rod worth is equal to the technical specification minimum shutdown margin. (model revision)

"Total Rod Worth

The total rod worth is set to the technical specification minimum shutdown margin, currently 1.0% $\Delta k/k$ converted to dollar units by diving by the value for beta effective."

Technical Justification: The simplification of the modeling of the reactivity insertion following reactor trip as described in the revision to "Fission Heat" described above, also simplifies the total rod worth assumed in the analysis. Since no thermal feedback is modeled, the total rod worth is simply set to the technical specification shutdown margin value, currently 1.0%Δk/k. This is converted to dollar units by dividing by the value for beta effective, the delayed neutron fraction. This modeling revision is only a simplification relative to the original methodology and does not change the results.

38. Section 4.3.1, p. 4-4, add new subsection "Fission Products and Actinides Decay" at the bottom of Section 4.3.1 to include a description of the decay heat modeling. This text was inadvertently not included in the original methodology, and is unchanged in Revision 1. The text from Section 3.3.1.1 on p. 3-10 is used without revision. (editorial)

    "Fission Product and Actinides Decay

    The heat produced from the radioactive decay of fission products and actinides is calculated for end-of-cycle conditions. This calculation determines decay heat as a function of time after trip for Oconee Units 1, 2, and 3 using the ANSI/ANS-5.1-1979 standard (Reference 3-5). For additional conservatism, uncertainties (2σ) are calculated and added to the mean decay heat values."

39. Section 4.3.2, p. 4-6, Reactor Building Model, is deleted in its entirety and replaced with a new subsection titled "Containment Boundary Conditions", since this model is no longer used in the Revision 1 methodology. (model revision)

    Delete: Section 4.3.2, Subsection "Reactor Building Model" in its entirety.

    Insert: New Subsection "Containment Boundary Conditions":

    "Containment Boundary Conditions

    Results from FATHOMS analyses are used as inputs to the RELAP5 analyses to model two containment boundary conditions. One boundary condition is the containment backpressure that the LOCA break flow discharges into. The second boundary condition is the containment sump water temperature used for the ECCS suction supply after the transition to the sump recirculation mode. These are selected based on FATHOMS cases that are consistent with the RELAP5 analysis in terms of break size and intended application."

    Technical Justification: Refer to the above Section 4.1, Analysis Overview, p. 4-2 revision technical justification for this related revision.

40. Section 4.3.2, pp. 4-7, 8, and 9, BWST Depletion - ECCS and Containment System Flow Rates, is revised in three places to no longer use RELAP5 for modeling the containment boundary conditions. FATHOMS results are used instead. (model revision)

    Change: "As discussed above, one of the primary purposes of the RELAP5 containment model is to provide an estimate of when this system is activated."

    To: "Containment analysis inputs required for the RELAP5 analysis are obtained from FATHOMS analysis results."

    Change: "The spilled HPI flow is accounted for in the RELAP5 containment model and the FATHOMS analysis as a flow boundary condition."

To: "The spilled HPI flow is modeled in the RELAP5 analysis and is used in the FATHOMS analysis as a flow boundary condition."

Change: "The RBS System is assumed to initiate when pressure in the RELAP5 containment model reaches 20 psig."

To: "The RBS System is assumed to initiate consistent with the FATHOMS results."

Technical Justification: Refer to the above Section 4.1, Analysis Overview, p. 4-2 revision technical justification for this related revision.

41. Section 4.3.2, p. 4-9, Steam Generator Level, is revised in the modeling of EFW actuation and control for consistency with the current station procedures and the UFSAR Chapter 15 SBLOCA peak cladding temperature analysis. (model revision)

    Change: "However, a ten minute delay from reactor trip is assumed in this analysis. Beyond 10 minutes, steam generator level is controlled at the LSCM setpoint minus instrument errors (56 inches) with the EFW System."

    To: "However, a 20 minute delay from reactor trip is assumed in the methodology. Furthermore, only one motor-driven EFW pump is credited to feed one steam generator to the LSCM setpoint minus an allowance for level instrument uncertainty."

42. Section 4.3.2, p. 4-10, Steam Generator Pressure Control, is revised to change the time at which the atmospheric steam dump valves are credited and the target cooldown rate. This change is necessary to be consistent with current station procedures. (model revision)

    Change: "An operator is assumed to begin manipulating these valves locally beginning 30 minutes after break initiation in an attempt to maintain a target cooldown rate of 80°F/hr, based upon the core exit temperature."

    To: An operator is assumed to begin manipulating these valves locally beginning 60 minutes after break initiation in an attempt to maintain a cooldown rate >50°F/hr, based upon the core exit temperature."

    Technical Justification: Current station procedures initiate the opening of the atmospheric steam dump valves to initiate the post-LOCA cooldown by no later than 60 minutes. The cooldown rate is targeted for the technical specification maximum of 100°F/hr. An actual cooldown rate of no less than 50°F/hr will be achieved. The revised methodology is updated to be consistent with the current station procedures.

43. Section 4.3.2, p. 4-10, Emergency Feedwater System, the modeling is revised to credit only one motor-driven emergency feedwater pump at 20 minutes. This change is necessary to be consistent with current station design and procedures. (model revision)

    Change: " The single failure of a 4160V switchgear will disable one motor-driven emergency feedwater (MDEFW) pump leaving one MDEFW pump and the turbine-driven emergency feedwater (TDEFW) pump. The operator is assumed to trip the one remaining MDEFW pump and control levels with the TDEFW pump feeding both steam generators. The maximum possible EFW flow is limited by the flow capacity of a single

TDEFW pump. A RELAP5 control system is used to control the actual EFW flow based upon the desired steam generator setpoint. The B&W EFW spreading model is used to model the phenomena of EFW spraying through the upper header onto the tubes in the outer periphery of the steam generators (see Section 2.1.1.2)."

To: " The single failure of a 4160V switchgear will disable one motor-driven emergency feedwater (MDEFW) pump, leaving one MDEFW pump and the turbine-driven emergency feedwater (TDEFW) pump available. No credit is taken for the TDEFW pump, leaving one MDEFW pump feeding one steam generator. A RELAP5 control system is used to control the actual EFW flow based upon the desired steam generator setpoint. The B&W EFW spreading model is used to model the phenomena of EFW spraying through the upper header onto the tubes in the outer periphery of the steam generators (see Section 2.1.1.2). Since MFW is conservatively used to feed to the natural circulation level setpoint, EFW is only used to raise the steam generator levels to the loss of subcooled margin setpoint. This is assumed to begin at 20 minutes based on manual operator action consistent with procedures. This is also consistent with the UFSAR Chapter 15 SBLOCA peak cladding temperature analysis."

Technical Justification: The original methodology for modeling EFW actuation and control for SBLOCA mass and energy release has been significantly revised to reflect current station operating procedures and for consistency with the SBLOCA peak cladding temperature analyses in UFSAR Chapter 15. This involves crediting only one MDEFW pump beginning at 20 minutes with flow delivered to only one steam generator. Furthermore, the EFW is only used to raise the steam generator level from the natural circulation setpoint to the loss of subcooled margin setpoint. This manual action is necessary to establish the boiler-condenser mode of heat transfer following SBLOCAs that require primary-to-secondary heat transfer for core cooling and unit cooldown.

44. Section 4.4.2, p. 4-14, third paragraph, third sentence, change "increase" to "decrease". (editorial)

Change: " This causes the void fraction in the break volume to increase which shifts the break flow from two-phase to single phase liquid."

To: " This causes the void fraction in the break volume to decrease which shifts the break flow from two-phase to single phase liquid."

45. Section 4.4.4, p. 4-18, first sentence, change "five times larger than" to "250% of". (editorial)

Change: " The break area assumed in this case is five times larger than the area assumed in the previous case."

To: " The break area assumed in this case is 250% the area assumed in the previous case."

46. Table 4.4-3, p. 4-25, change "Case 2" and "Case 3" to "Case 5" and "Case 6" in the table headings, respectively. (editorial)

Change: "Case 2"

To: "Case 5"

Change: "Case 3"

To: "Case 6"

Chapter 5

47. Section 5.1.1, Description of Steam Line Break Accident, p. 5-1, change "UFSAR Section 15.13" to "UFSAR Section 6.2 (Reference 5-1)" for consistency with the current UFSAR section for this content. (editorial)

    Change: "UFSAR Section 15.13"

    To: "UFSAR Section 6.2 (Reference 5-1)"

48. Section 5.1.2, Acceptance Criteria, p. 5-1, revise the text to simplify the acceptance criteria discussion so that it is applicable for the steam line break mass and energy release and containment response analysis. (editorial)

    Change: "The criteria for unit protection and the release of fission products to the environment for the steam line break accident as listed in UFSAR Section 15.13 (Reference 5-1) are as follows:

    1. The core will remain intact for effective core cooling, assuming minimum tripped rod worth with a stuck rod.
    2. No steam generator tube loss of primary boundary integrity will occur due to the loss of secondary side pressure and resultant temperature gradients.
    3. Doses will be within 10CFR100 limits.

    The stated UFSAR acceptance criteria for steam line break do not include containment design pressure as an acceptance criterion. However, a containment pressure response analysis was performed and is presented in the UFSAR with a comparison of the results to the design pressure limit. In addition, IE Bulletin 80-04 (Reference 5-2) requested a review of the containment pressure response analysis to determine the impact of runout flow from the Emergency Feedwater System and the impact of other energy sources, such as continuation of feedwater or condensate flow. Thus, containment pressure must remain below the design limit of 59 psig during a steam line break accident."
    "

    To: "The acceptance criteria for the steam line break mass and energy release and containment response analyses are that the containment pressure will not exceed the design pressure of 59 psig, and that safety-related equipment inside containment will survive the harsh environment resulting from the transient (equipment qualification)."

49. Section 5.4, p. 5-8, Main Steam Lines, change "Technical Specification 4.8" to "Technical Specification 3.7.2" for consistency with the current technical specifications. (editorial)

    Change: " Technical Specification 4.8"

To: " Technical Specification 3.7.2"

50. Section 5.4, p. 5-8, Main Feedwater Lines, change subsection title to "Main Feedwater System" to more accurately reflect the content. (editorial)

    Change: " Main Feedwater Lines"

    To: "Main Feedwater System"

51. Section 5.5 , p. 5-14, Offsite Power Maintained, second paragraph, change "affected loop saturates" to "unaffected loop saturates" to correct an error. (editorial)

    Change: "Reactor coolant pressure (Figure 5.5-1) rapidly decreases after the steam line break occurs until the affected loop saturates at about 10 seconds."

    To: " Reactor coolant pressure (Figure 5.5-1) rapidly decreases after the steam line break occurs until the unaffected loop saturates at about 10 seconds "

52. Section 5.6, References: One reference deleted and two references revised. (editorial)

    Reference 5-2 - Deleted

    Reference 5-5 - Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, May 2002

    Reference 5-7 - Oconee Nuclear Station Technical Specifications

    To: " Technical Specification 3.7.2"

Chapter 6

53. Section 6.1 , p. 6-1, Overview, second paragraph, the methodology is revised due to the large break LOCA and small break LOCA analyses being divided into two time segments instead of three. (model revision)

    Change: "For the long-term large break LOCA, three segments are used to achieve the best accuracy in the [ ] code. The small-break LOCA simulations are also divided into three segments . . ."

    To: "For the long-term large break LOCA, two segments are used to achieve the best accuracy in the [ ] code. The small-break LOCA simulations are also divided into two segments . . ."

    Technical Justification: The FATHOMS modeling is divided into time segments with common post-LOCA plant operating configurations, and therefore can be modeled with the same boundary conditions. Based on current modeling, the long-term large break LOCA can be modeled with two time segments instead of three time segments as stated in the original methodology. The small-break LOCA can now be modeled with two time segments instead of three also.

54. Section 6.2.3, Droplet Size of Blowdown Mass, p. 6-4, is revised in Revision 1 to change the droplet size to 100 microns and a new reference is provided. (model revision)

    Change: "A 20 μm value for the size of the blowdown droplets is used. Test data cited in Reference 6-3 suggests that the droplet sizes for high-pressure blowdowns range from 2 - 2000 μm, and Reference 6-4 suggests that 20 μm is a suitable average value to use with the FATHOMS code."

    To: "A 100 μm value for the size of the blowdown droplets is used in the Revision 1 methodology. Test data cited in Reference 6-3 suggests that the droplet sizes for high-pressure blowdowns range from 2 - 2000 μm, and Reference 6-8, suggests that 100 μm is a suitable average value to use with the FATHOMS code. A value of 20 μm value was used in the Revision 0 methodology based on Reference 6-4."

    Technical Justification: Reference 6-8 recommends use of a droplet size of 100 μm in the FATHOMS code. This change is made to update the methodology to a more current value.

55. Section 6.3.1, Overview, p. 6-11, top paragraph, the RELAP5 analysis now runs through to the end of the injection phase instead of stopping at 1800 seconds. (model revision)

    Change: ". . . only lasts for 1800 seconds, the BFLOW code is used to determine the quality of the flow exiting the break following the first 1800 seconds."

    To: ". . . only lasts until the start of the recirculation mode, the BFLOW code is used to determine the quality of the flow exiting the break thereafter."

    Technical Justification: In the Revision 0 methodology the RELAP5 analysis was stopped at 1800 seconds. In the Revision 1 methodology it was decided that a more appropriate time for terminating the RELAP5 analysis was at the end of the injection mode (beginning of the recirculation mode). This choice is a more natural time for transitioning from RELAP5 to the BFLOW code, since there is a step change in the ECCS suction water temperature at that time.

56. Section 6.3.2, Analytical Approach, p. 6-12, top paragraph, the RELAP5 analysis now runs through to the end of the injection phase instead of stopping at 1800 seconds. Also, there are two time segments instead of three. (model revision)

    Change: "Three different sets of flowpaths (junctions) are used in the analysis to implement the mass and energy release data supplied from the RELAP5 analysis and the data from the BFLOW code. Each set of flowpaths is active during one time segment of the analysis. The first segment ends at 1800 seconds, which is the end of the mass and energy release data from the RELAP5 analysis. The second segment lasts from 1800 seconds to 2808 seconds, which is the switchover point to sump recirculation mode. The third segment lasts from 2808 seconds to the end of the analysis, at 15 days. These segments will be designated as Segments 1, 2, and 3."

To: " Two different sets of flowpaths (junctions) are used in the analysis to implement the mass and energy release data supplied from the RELAP5 analysis and the data from the BFLOW code. Each set of flowpaths is active during one time segment of the analysis. The first segment ends at the end of the injection mode, which is the end of the mass and energy release data from the RELAP5 analysis. The second segment lasts from the start of the recirculation mode to the end of the analysis. These segments will be designated as Segments 1 and 2."

Technical Justification: In the Revision 0 methodology the RELAP5 analysis was stopped at 1800 seconds. In the Revision 1 methodology it was decided that a more appropriate time for terminating the RELAP5 analysis was at the end of the injection mode (beginning of the recirculation mode). This choice is a more natural time for transitioning from RELAP5 to the BFLOW code, since there is a step change in the ECCS suction water temperature at that time.

57. Section 6.3.3 , p. 6-14, BWST Volume and Temperature, the volume modeling is revised due to the large break LOCA analysis being divided into two time segments instead of three. The Revision 1 methodology is inserted in front of the Revision 0 methodology, and the Revision 0 text is only retained for historical purposes. (model revision)

    Insert the following new text at the beginning of this subsection:

    "During Segment 1, the ECCS injection flow coming from the BWST and the flow exiting the break are modeled in the RELAP5 analysis. After the switchover to sump recirculation mode (during Segment 2), the source for the ECCS water will be the sump, so the BWST is not modeled.

    Note: The following two paragraphs are superceded due to being applicable only to Revision 0 of the report. They are retained for comprehension of the Revision 0 results only."

    Technical Justification: The FATHOMS modeling is divided into time segments with common post-LOCA plant operating configurations, and therefore each time segment can be modeled with the same boundary conditions. Based on current experience, the long-term large break LOCA can be modeled with two time segments instead of three time segments as stated in the original methodology. The original methodology is retained only so that the results presented can be understood.

58. Section 6.3.3 , p. 6-15, Heat Transfer Correlation, first paragraph, delete "1800 seconds". Also, in the second paragraph, the number of time segments is changed from 3 to 2. (model revision)

    Change: ". . . start of Segment 2 (1800 seconds) . . . "

    To: ". . . start of Segment 2 . . ."

    Change: "During Segments 2 and 3, the sump region is over . . . "

    To: "During Segment 2 the sump region is over . . . "

Technical Justification: The FATHOMS modeling is divided into time segments with common post-LOCA plant operating configurations, and therefore can be modeled with the same boundary conditions. Based on current modeling, the long-term large break LOCA can be modeled with two time segments instead of three time segments as stated in the original methodology. In the Revision 1 methodology the end of Segment 1 is at the end of the injection mode, rather than at 1800 seconds. This choice is a more natural time for transitioning from RELAP5 to the BFLOW code, since there is a step change in the ECCS suction water temperature at that time.

59. Section 6.3.3 , p. 6-15, Group 2 and 3 Heat Structures, second paragraph, the number of time segments is changed from 3 to 2. (model revision)

Change: "A constant heat transfer coefficient is assumed for these heat structures during Segments 2 and 3."

To: " A constant heat transfer coefficient is assumed for these heat structures during Segment 2."

Technical Justification: The FATHOMS modeling is divided into time segments with common post-LOCA plant operating configurations, and therefore can be modeled with the same boundary conditions. Based on current modeling, the long-term large break LOCA can be modeled with two time segments instead of three time segments as stated in the original methodology.

60. Section 6.3.3, Droplet Size of Blowdown Mass, p. 6-16, is revised in Revision 1 to change the droplet size to 100 microns. (model revision)

Change: "A 20 μm value for the size of the blowdown droplets is again used."

To: "A 100 μm value for the size of the blowdown droplets is again used."

Technical Justification: Reference 6-8 recommends use of a droplet size of 100 μm in the FATHOMS code. This change is made to update the methodology to a more current value.

61. Section 6.3.3, Modifications to Base Containment Model, p. 6-16, insert a new subsection titled "LPI Flowpaths and Mixing Volume" prior to subsection "LPI Cooler Data". (model revision)

Insert the following new text:

"LPI Flowpaths and Mixing Volume The LPI System has two trains with two separate LPI coolers. For some failure scenarios it is possible for only one cooler to be supplied with cooling water, although both will have LPI flow. To model this asymmetric configuration, two LPI flowpaths from the sump are modeled with separate LPI coolers. Downstream of the LPI coolers the two flows are then combined in a mixing volume with one outlet flowpath."

Technical Justification: The Revision 0 model combined both LPI trains into one equivalent train for modeling purposes. This was thought to be an adequate model for the intended applications. Subsequently, it was determined that some failure scenarios existed for which cooling water would not be supplied to one of the two LPI coolers. However, both LPI trains would still have LPI flow. This would result in one train of cooled LPI flow, and one train of LPI pumping sump water back into the reactor vessel without cooling. This is accommodated in the Revision 1 model by separately modeling both LPI trains from the sump to the LPI cooler outlet, and then combining the flows exiting the LPI coolers into a mixing volume. This mixing volume is non-physical since the two trains are not necessarily headered in the actual valve alignment. However, this modeling approach is functionally correct for the intended simulation purposes and is used due to the modeling of the long-term decay heat in the FATHOMS model.

62. Section 6.3.5, Boundary Conditions, p. 6-19, a new table of junctions and boundary conditions is inserted for Revision 1, and the existing table is noted as Revision 0. This table is only a summary of revisions explained in other items. (editorial)

Insert the following new table and text prior to the existing table:

1.	Cold leg break - side #1 (vap)	Seg. 1	Injection mode
2.	Cold leg break - side #1 (liq)	Seg. 1	Injection mode
3.	Cold leg break - side #2 (vap)	Seg. 1	Injection mode
4.	Cold leg break - side #2 (liq)	Seg. 1	Injection mode
10.	Sump drain - to LPI cooler #1	Seg. 2	Recirculation mode
11.	LPI cooler #1 - to mixing volume	Seg. 2	Recirculation mode
12.	Sump drain - to LPI cooler #2	Seg. 2	Recirculation mode
13.	LPI cooler #2 - to mixing volume	Seg. 2	Recirculation mode
14.	Sump drain - to Building Spray	Seg. 2	Recirculation mode
15.	Building Spray flow - from sump	Seg. 2	Recirculation mode
16.	Building Spray flow - initial	Seg. 1	Injection mode
17.	Nitrogen injection from CFTs	Seg. 1	Injection mode
19.	From mixing volume - to core	Seg. 2	Recirculation mode
20.	From core - to containment (vapor phase)	Seg. 2	Recirculation mode
21.	From core - to containment (liquid phase)	Seg. 2	Recirculation mode

"Note: The following table is applicable to the Revision 0 FATHOMS model only, and is included so that the results of the Revision 0 analysis can be understood."

63. Section 6.3.5.1, LPI/BS Discrepancy Flowpath (18), p. 6-21, can be deleted since it is no longer part of the methodology. (model revision)

Delete: Subsection "LPI/BS Discrepancy Flowpath"

Technical Justification: The LPI/BS discrepancy flowpath was part of the original methodology based on the transition from RELAP5 to FATHOMS. This approach to address an inconsistency is no longer applicable or necessary with Revision 1. It is therefore deleted.

64. Section 6.3.5.2, Segment 2 Flow Paths, pp. 6-21 to 6-23 can be deleted since Segment 2 is no longer part of the methodology. Section 6.3.5.3 will be resequenced. (model revision)

    Delete: The text of Section 6.3.5.2 "Segment 2 Flow Paths" except for the title.

    Technical Justification: Segment 2 was part of the original methodology but is not part of the Revision 1 methodology. Therefore the text can be deleted. The title is retained with new text from the subsequent section (see below).

65. Section 6.3.5.3 (actually 6.5.4.3 in the report due to an error), Segment 3 Flow Paths, p. 6-23, delete the section title and move the text to Section 6.3.5.2 to reflect the deletion of the old Segment 2. The dual LPI flowpaths and the mixing volume are added. Text discussing the Reactor Building Spray System is no longer applicable and is deleted. (model revision)

    Delete: The title of Section 6.3.5.3 (actually 6.5.4.3 in the report due to an error)

    Change: " LPI Flowpaths (10-13) At 2808 seconds, the recirculation phase is initiated. Water is taken from the sump drain through boundary condition 10, cooled by the LPI coolers, and then injected into the core. As during Segment 2, the quality matrices from the BFLOW analysis are used to determine the quality of the flow exiting the break. This exit flow is again separated by phase through separate boundary conditions (11 and 12). Also as in Segment 2, the LPI flow rate is reduced by the instrument uncertainty value for conservatism. Boundary condition 13, which takes water from the LPI cooler outlet to Building Spray, is not used; all Building Spray comes directly from the sump during recirculation for this analysis. However, the option of using the LPI coolers for cooling Building Spray water is still available at Oconee."

    To: "LPI Flowpaths (10-13, 19-21) At the start of the recirculation phase water is taken from the sump drain through boundary conditions 10 and 12, and then cooled by the two LPI coolers. Downstream of the LPI coolers the two LPI flows are then combined in a mixing volume via boundary conditions 11 and 13, and then the flow is directed to the core via boundary conditions 19-21. The[ ]BFLOW analysis are used to determine the quality of the flow exiting the break. This exit flow is again separated by phase through separate boundary conditions (20 and 21). The LPI flow rate is reduced by the instrument uncertainty value for conservatism."

    Technical Justification: Revision 0 included only one equivalent LPI train, which did not allow for asymmetric LPI train performance. The Revision 1 model has been changed to allow both LPI trains to be modeled from the sump to the LPI cooler outlet. At that point the two trains are combined into a mixing volume before returning the flow to the core. Various flowpaths and boundary conditions are changed to make these modeling revisions. Also, the modeling of cooled spray water via the LPI coolers is now longer an option at Oconee and has been deleted from the model.

66. Section 6.5.4.3 , Building Spray (14-15), p. 6-23, first paragraph, the number of time segments is changed from 3 to 2. (model revision)

Change: "Another set of boundary conditions models the BS flowpaths during Segment 3."

To: "Another set of boundary conditions models the BS flowpaths during Segment 2."

Technical Justification: The FATHOMS modeling is divided into time segments with common post-LOCA plant operating configurations, and therefore can be modeled with the same boundary conditions. Based on current modeling, the long-term large break LOCA can be modeled with two time segments instead of three time segments as stated in the original methodology.

67. Section 6.4.2, Analytical Approach, p. 6-29, first paragraph, deleted the boron dilution line since its use for this purpose has been removed from the methodology due to design issues. (plant design change)

    Delete: "The boron dilution line is also modeled for Cases 3 and 4. As described in Section 2.2.1.3, the boron dilution line allows saturated liquid to exit the reactor vessel to the Reactor Building sump. Whenever this flow path is opened, the size of the RCS break is effectively increased. Taking credit for this flow path from the RCS to the Reactor Building sump allows more flow to exit the RCS, therefore allowing more cool water to be injected into the RCS and aid in the system cooldown."

    Technical Basis: The use of the boron dilution line for this purpose has been removed from station procedures due to design issues.

68. Section 6.4.3, BWST Volume, p. 6-30, is not included in the Revision 1 methodology, since the BWST is no longer modeled in the FATHOMS analysis. Text is inserted in front of the Revision 0 methodology to discuss this change, and the Revision 0 text is only retained for historical purposes. (model revision)

    Insert the following new text at the beginning of this subsection:

    "During Segment 1, the ECCS flow coming from the BWST and the flow exiting the break are modeled in the RELAP5 analysis. After the switchover to sump recirculation mode (during Segment 2), the source for the ECCS injection water will be the sump, so the BWST is not modeled.

    Note: The following paragraph is superceded due to being applicable only to Revision 0 of the report. The text is retained for comprehension of the Revision 0 results only."

    Technical Basis: In the original methodology the RELAP5 analysis would be terminated and a transition to FATHOMS would be made to continue the analysis. This was based on time at 30 minutes, and may be with the ECCS still aligned for injection from the BWST. With the Revision 1 methodology the ECCS will be aligned to the sump at the time of the transition to FATHOMS. Therefore, modeling the BWST in FATHOMS is deleted from the methodology.

69. Section 6.4.3, Junctions Descriptions, p. 6-31, the second paragraph is revised to indicate that opening the boron dilution line for this purpose has been deleted in Revision 1 from the

methodology due to design issues. The existing text is retained for historical purposes.(plant design change)

Insert the following new paragraph before the second paragraph on p. 6-31:

"Note: The boron dilution line is no longer opened in the Revision 1 methodology. The following paragraph regarding Revision 0 results is retained for historical purposes."

Technical Basis: The use of the boron dilution line has been removed from the methodology in Revision 1 due to design issues associated with its use for this purpose. This change is consistent with current station procedures.

70. Section 6.4.3, RCS Heat Structures, p. 6-31, is re-titled "RCS and Secondary Heat Structures" and revisions are made to include the modeling of steam generator secondary side stored energy. (model revision)

Change:

<u>"RCS Heat Structures</u>

[ ]

To:

<u>"RCS and Secondary Heat Structures</u>

[ ]

[ ]

Technical Justification:

[ ]

71. Section 6.4.3, Droplet Size of Blowdown Mass, p. 6-32, is revised in Revision 1 to change the droplet size to 100 microns. (model revision)

    Change: "A 20 μm value for the size of the blowdown droplets is again used."

    To: "A 100 μm value for the size of the blowdown droplets is again used."

    Technical Justification: Reference 6-8 recommends use of a droplet size of 100 μm in the FATHOMS code. This change is made to update the methodology to a more current value.

72. Section 6.4.5, Boundary Condition, p. 6-33, is revised to indicate that the Revision 1 methodology has 14 instead of 17 junctions. A statement is added to identify which three are no longer in the methodology. The Revision 0 text is retained for the three deleted junctions for historical purposes. (model revision)

    Change: "There are 17 junctions in the model, with all but three assigned to a particular boundary condition."

    To: "There are 14 junctions in the Revision 1 model, whereas there were 17 junctions in the Revision 0 model. Junctions 10, 11, and 17 have been deleted from the methodology. The description of these three deleted junctions in the table and text that follows is retained so that the Revision 0 results can be understood." Also, Junction 10,11, and 17 have an asterisk added and the following footnote is inserted at the bottom of the table on p. 6-33. "Note: "*" indicates this junction is deleted in Revision 1".

    Technical Justification: Junctions 10, 11, and 17 have been deleted from the methodology for two reasons. Junctions 10 and 11 have been deleted since the transition from RELAP5 to BFLOW has been shifted until after the alignment for sump recirculation. Therefore these two junctions are no longer applicable. Junction 17 has been deleted since the use of the boron dilution line for this purpose has been removed due to station design issues.

73. Section 6.4.5, HPI Spill (before recirculation) (8), p. 6-34, is revised to delete the second sentence regarding the FATHOMS BWST model, which has been deleted in Revision 1. (model revision)

Delete: " The inventory from the BWST is compensated for in the RELAP5 analyses, so this water is not drawn from the FATHOMS BWST model until the RELAP5 mass and energy release data is exhausted.."

Technical Justification: The BWST is no longer modeled in FATHOMS in the Revision 1 methodology. The remaining text with this revision correctly describes the current methodology.

74. Section 6.6, References: Reference 6-2 corrected and new Reference 6-8 added. (editorial)

Reference 6-2 - NUREG-0588, Interim Staff Position on Environment Qualification of Safety-Related Electrical Equipment, U.S. Nuclear Regulatory Commission, December 1979

Reference 6-8 - "Drop Behavior in GOTHIC," NAI-9301-7, Numerical Application, Inc., January 10, 2001

Chapter 7

75. Chapter 7, p. 7-1 is revised to add the following paragraphs at the end to summarize the content of the supplements and appendices. (editorial)

"Supplement 1 was included in the November 1997 publication of the report to provide the results of an application of the methodology with the Main Steam Line Break Detection and Main Feedwater Isolation System. This system was installed subsequent to the approval of the original topical report, and significantly changes the results by crediting automatic isolation of MFW and partial isolation of EFW. This application of the methodology was provided since it superceded the application in Chapters 5 and 6 of Revision 0.

Supplement 2 was added with the November 1997 version of Revision 0 of the report to present validation analyses comparing the FATHOMS prediction to data from the CVTR facility.

Appendix A was added in Revision 1 to describe the replacement steam generator (ROTSG) steam line break mass and energy release modeling with RETRAN-3D.

Appendix B was added in Revision 1 to address the RETRAN-3D SER conditions and limitations as relates to the replacement steam generator (ROTSG) steam line break mass and energy release modeling with RETRAN-3D.

Appendix C was added in Revision 1 to include the use of the GOTHIC 7.0 code for containment response analysis."

Supplement 1

76. Subsection RETRAN-02 System Analysis, p. S1-4, first paragraph, change "affected loop saturates" to "unaffected loop saturates" to correct an error. (editorial)

    Change: "Reactor coolant pressure (Figure S1-1) rapidly decreases after the steam line break occurs until the affected loop saturates at about 10 seconds."

    To: " Reactor coolant pressure (Figure S1-1) rapidly decreases after the steam line break occurs until the unaffected loop saturates at about 10 seconds "

APPENDIX A

METHODOLOGY REVISION FOR OCONEE REPLACEMENT STEAM GENERATORS

ROTSG Design Description

This appendix describes the Oconee RETRAN model revisions necessary for simulation of the steam line break mass and energy release analysis for the replacement steam generators (ROTSGs) being manufactured by Babcock & Wilcox Canada (BWC). The first pair of ROTSGs are scheduled for installation in Oconee Unit 1 in 2003. The other two Oconee units are scheduled for replacement outages in 2004. The design of the ROTSGs enables what is characterized as a "like-for-like" replacement due to the close similarity in the performance of the component. Due to the similarity of the ROTSGs with respect to the original OTSGs, the RETRAN modeling is also maintained very similar. The new RETRAN models for the ROTSGs are detailed in Revision 3 to Duke Power topical report DPC-NE-3003, "Thermal-Hydraulic Transient Analysis Methodology" (Reference A-1). Additional new modeling details specific to the ROTSG steam line break mass and energy release analysis are included in this appendix. The significant design differences are as follows. Figure A-1 shows a comparison of the OTSG and ROTSG designs.

1) Flow-restricting orifices in the steam outlet nozzles
2) Inconel-690 tubes
3) 15,631 vs 15,531 tubes
4) Thinner pressure vessel / wider downcomer
5) Thinner tubesheets resulting in 3.625 inch longer heated tube length
6) 1.2% greater heat transfer area
7) More water in the steam generator.

Other than the ROTSG dimensional design data, Duke is also using steady-state thermal-hydraulic data provided by BWC as the reference data for the ROTSG initial conditions. These data included the pressure distribution, void fraction distribution, steam superheat profile, and water masses. The BWC simulations are three-dimensional, whereas RETRAN predictions are one-dimensional, so some approximation is necessary when comparing the results of the two codes.

RETRAN-3D Code

The Electric Power Research Institute's (EPRI) RETRAN-3D code (Reference A-2) is used for the Oconee RETRAN Model with ROTSGs. Previous modeling described in the body of this report with the original OTSGs used the EPRI RETRAN-02 code. RETRAN-3D, was approved by the NRC in the SER dated January 25, 2001 (Reference A-3). The most significant change in the Oconee RETRAN methodology resulting from this RETRAN code version change is in the two-phase modeling of the secondary side of the steam generators. With RETRAN-02 the secondary side is modeled using the homogeneous-equilibrium-mixture (HEM) equations. This was necessary due to initialization limitations that prevented initialization of OTSGs with RETRAN-02's two-phase modeling options. With RETRAN-3D the secondary side is modeled using the Chexal-Lellouche algebraic slip drift flux model. This is the vendor-recommended slip option and the option reviewed in detail by the NRC. The application of this model has been modified by CSA and Duke to enable benchmarking the ROTSG secondary side water distribution predicted by BWC's design code. All of the details of the transition to using RETRAN-3D instead of RETRAN-02 for Oconee modeling are presented. It is noted that the three-dimensional core model capability in RETRAN-3D is not included in this methodology, as well as some of the other new models. The conditions and limitations in the NRC's SER for RETRAN-3D are addressed in Appendix C.

ROTSG Nodalization

The ROTSG nodalization is shown in Figure A-2. This nodalization can be compared to the OTSG nodalization in Reference A-1, Figure 2.2-1.

[ ]

Details of RETRAN-3D Modeling

The selection of RETRAN-3D models, code options, and other input specification for the Oconee RETRAN Model with ROTSGs is presented and justified in Reference A-1, Appendix A. RETRAN-3D code modifications developed by Computer Simulation & Analysis, Inc, for Duke are also presented in detail in Reference A-1, Appendix A. The content of Reference A-1, Appendix A is not repeated here, but is incorporated in the methodology by reference. Additional methodology revisions specific to the steam line break mass and energy release analysis follow.

Revisions to Chapter 5 Methodology

Section 5.2.2 - Break Modeling
A spectrum of break sizes is analyzed in the Revision 1 methodology.

Technical Justification: The original methodology, although not excluding other than the double-ended maximum break size, did not investigate smaller break sizes. The application of the revised methodology covers a range of break sizes. This was determined to be necessary due to the ROTSG steam outlet nozzle flow orifices and the new Automatic Feedwater Isolation System (AFIS) station design modification.

Section 5.3 - Initial Conditions
Although initial conditions are considered inputs to the methodology rather than methodology, many of the initial conditions in Section 5.3 are not applicable to the ROTSG application. Changes to input values are not included.

Technical Justification: Changes to input values are not methodology changes and therefore do not need to be revised.

Section 5.4 Boundary Conditions - Main Feedwater System
This section is supplemented in Revision 1 by the Integrated Control System (ICS) and Automatic Feedwater Isolation System (AFIS) discussion below for additional modeling details regarding main feedwater control.

Section 5.4 Boundary Conditions - Emergency Feedwater System
This section is supplemented in Revision 1 by the Automatic Feedwater Isolation System (AFIS) discussion below for additional modeling details regarding emergency feedwater control.

Section 5.4 Boundary Conditions - Single Failure
This section is supplemented in Revision 1 to state that single failures are modeled in a consistent manner between the RETRAN-3D mass and energy release analysis and the FATHOMS or GOTHIC containment response analysis. For example, a single failure of a 4160V switchgear will have consistent results on HPI, EFW and RBS pumps and valves.

Technical Justification: Traditional transient and accident analyses assume a single failure in safety-related components or systems. This methodology clarification states that the single failure will be consistently applied to both the RETRAN and the FATHOMS or GOTHIC analysis. In other words, the consequences of the single failure will be realistic in terms of what components fail due to that specific single failure.

Section 5.4 Boundary Conditions - Reactor Protective System Trips
For the analysis of some smaller break sizes reactor trip on high Reactor Building pressure is credited in Revision 1. An uncertainty of 0.37 psi is added to the technical specification allowable value of 4.0 psig to ensure a late reactor trip. A conservatively long delay time of 0.4 seconds is assumed for this trip function. The time at which the RETRAN-3D analyses assume reactor trip on high Reactor Building pressure is dependent on break size and is determined by iteration with the FATHOMS or GOTHIC results.

Technical Justification: In the original methodology the focus was on double-ended maximum break sizes. In the application of the methodology to the ROTSGs a break spectrum was investigated. For some of the smaller break sizes the reactor trip on primary system trip functions such as variable low pressure-temperature (P/T) was delayed. A trip on high Reactor Building pressure would occur earlier than the P/T trip. Credit for the high Reactor Building pressure trip is included in the Revision 1 methodology. The FATHOMS or GOTHIC code is used to conservatively predict the timing of the reactor trip for each break size and set of analysis assumptions. A conservative high Reactor Building pressure trip setpoint including uncertainty, and a bounding trip delay time are included.

Section 5.4 Boundary Conditions - Integrated Control System (ICS)
The description of the modeling of the Integrated Control System (ICS) is expanded in Revision 1 to address the control of the Main Feedwater System (MFW) when the ICS is

initially in the manual control mode. The ICS switches to the automatic control mode upon reactor trip, and controls level to the appropriate post-trip level setpoint.

Technical Justification: Normally the ICS is in the automatic control mode. The predominant reason for the ICS to be in the manual control mode is if there has been some component failure in the ICS, or some other off-normal operational restriction exists. Repairs to the ICS to allow an expeditious return to the automatic control mode would be a high priority and would be completed in a timely manner. Therefore, operating with the ICS in manual is very infrequent and would be of short duration. With the ICS in manual the operations staff is highly focused on their responsibility for manual control. With the ICS in manual at the initiation of a transient, the MFW System will respond to the change in steam generator backpressure. If pressure decreases the MFW pump will shift along its head/flow curve and flow will increase. This is what will occur following a steam line break event. All steam line break events inside containment will cause a rapid pressurization of the Reactor Building and a reactor trip on high pressure at 4.0 psig plus uncertainty. In response to the reactor trip signal, the ICS will automatically switch from manual control to auto control, and will perform its normal post-trip design function of controlling steam generator level to setpoint. The setpoint is 25 inches on the startup level range with any reactor coolant pumps in operation, or 50% on the operating range level if all reactor coolant pumps are tripped. An allowance is made for the effect of a harsh containment environment on the steam generator level control setpoint. This ICS response is modeled in the steam line break analysis with the ICS initially in the manual control mode. A failure of the ICS to respond in this manner would constitute a failure in a control system. Failures in control systems are not assumed in conjunction with UFSAR transients and accidents in Oconee's licensing basis.

<u>Section 5.4 Boundary Conditions - Liquid Carryout and Superheat Modeling</u>
The original methodology for addressing the consequences of liquid carryout following a steam line break is revised in Revision 1. This modeling approach is necessary to conservatively correct the RETRAN-3D results that include liquid carryout since the carryout prediction cannot be validated due to an absence of data. Correcting for carryout is necessary to avoid potentially under-predicting the energy release. Computer Simulation & Applications, Inc., the RETRAN vendor, and Duke Power jointly developed the Revision 1 methodology that follows.

[ ]

<u>Card 08XXXY - Junction Data</u>

Section 5.4 Boundary Conditions - Automatic Feedwater Isolation System (AFIS)

AFIS is designed to isolate MFW and EFW to a faulted steam generator during a steam line break event, thereby eliminating the need for prompt manual operator action to isolate EFW to a faulted steam generator. Detection of the steam line break event is accomplished by monitoring four pressure transmitters on each of the two main steam lines. When main steam pressure falls below the low pressure setpoint (currently, 520 psig including a 30 psi uncertainty), the MFW pumps are tripped and the turbine-driven EFW pump is stopped. In addition, the MFW block valves, control valves, startup block valves, and startup control valves on the depressurized steam generator are closed. When a high rate of depressurization is sensed (currently, -3.3 psi/sec including a 0.3 psi/sec uncertainty) and pressure is below the low pressure setpoint, the motor-driven EFW pump(s) is tripped on the depressurized header(s). For break sizes where the rate of depressurization setpoint is not exceeded, manual operator action is credited at 10 minutes to isolate motor-driven EFW flow to the affected steam generator. Conservative modeling of the response of the components affected by an AFIS actuation is credited in the steam line break mass and energy release analyses.

Technical Justification: The above description of the AFIS design is accurately modeled in the RETRAN-3D ROTSG steam line break. This new plant modification is credited with automatically isolating MFW and EFW to a depressurizing steam generator. The AFIS design will replace the Main Steam Line Break Detection and Feedwater Isolation System design presented in Supplement 1.

References

A-1 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002.

A-2 RETRAN-3D- A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI, NP-7450(A), Volumes 1-4, Revision 5, July 2001

A-3 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

Figure A-1

OTSG / ROTSG Comparison

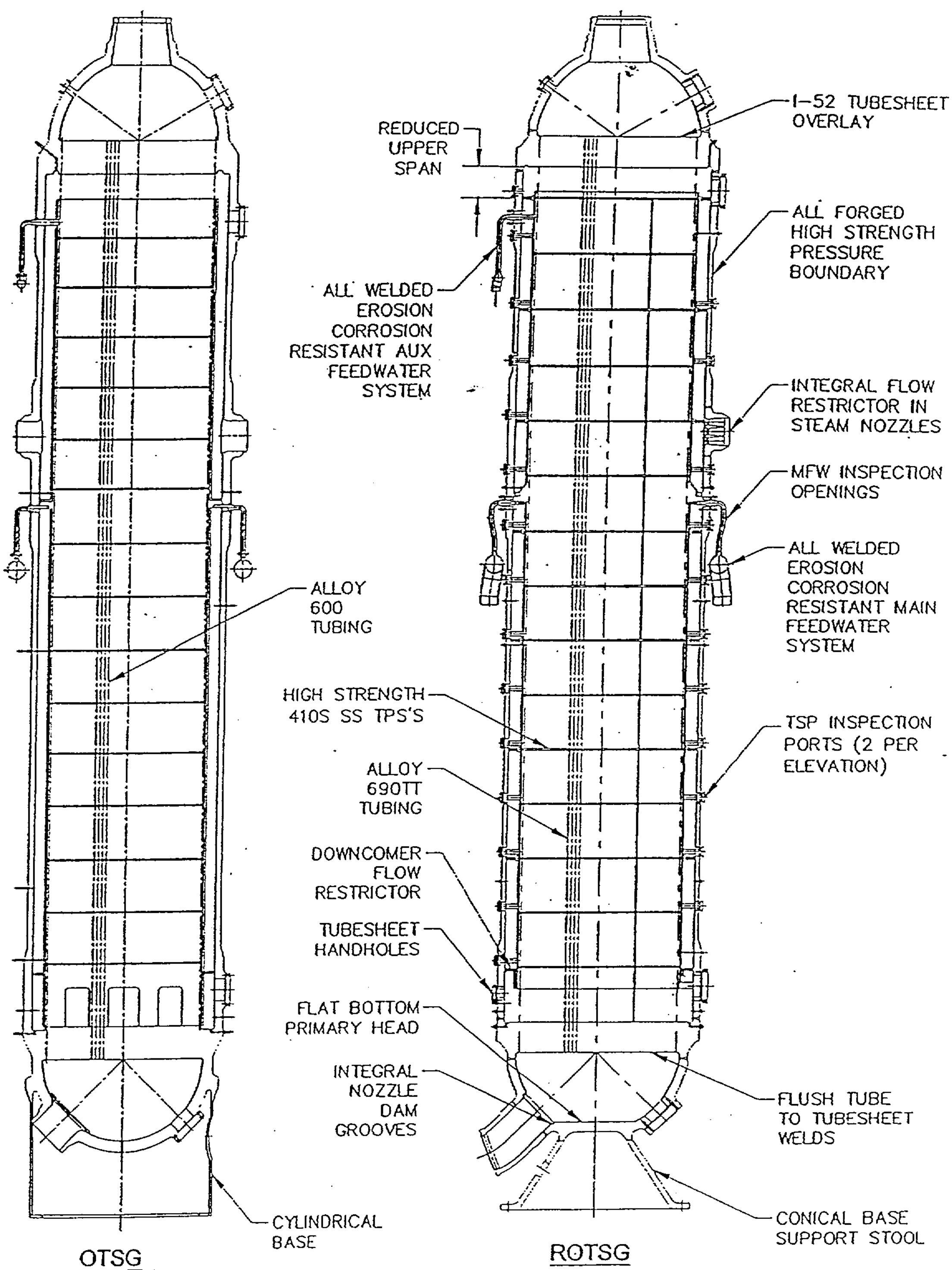


Figure A-2

ROTSG Nodalization

APPENDIX B

EVALUATION OF RETRAN-3D SER CONDITIONS AND LIMITATIONS FOR THE OCONEE RETRAN MODEL WITH ROTSGs

Purpose

This appendix evaluates the conditions and limitations in the RETRAN-3D SER (Reference B-1) for the application of RETRAN-3D to the Oconee Nuclear Station with replacement steam generators (ROTSGs). The results of this evaluation demonstrate that the use of the RETRAN-3D code for this application, as described in Appendix B, has been appropriately justified and is within the SER conditions and limitations. Therefore, the approval for use of RETRAN-3D as stated in the SER for the scope of approval specified in the SER can be credited. The Duke version of the RETRAN-3D code actually used is described first, including details on the custom code modifications that have been incorporated.

Description of the RETRAN-3D Code Version Used and Duke Code Modifications

RETRAN-3D MOD003.1DKE is the current Duke Power version of the standard Electric Power Research Institute (EPRI) RETRAN-3D MOD003.1 code, the NRC-approved code. This EPRI version includes the revisions agreed to during the SER review process. The Duke Power version, designated by the suffix "DKE", consists of two types of revisions to the standard EPRI version. The first type of revision is code error corrections. Duke periodically updates the code version in use to include error corrections obtained from Computer Simulation & Analysis, Inc. (CSA), the EPRI contractor for the RETRAN-3D code. The second type of revision is Duke custom code modifications purchased from CSA to address Duke-specific modeling needs. Each of these custom code changes are described in detail for NRC review and approval. The error corrections are not presented. All error corrections and code modifications are developed and controlled under CSA's Appendix B QA program. All Duke RETRAN-3D versions used for safety-related applications are certified and controlled under Duke's Appendix B software quality assurance program (Reference B-2). Any future modifications to the RETRAN-3D code versions used by Duke for safety-related applications that constitute significant model revisions or new models will be submitted for NRC review and approval. Code modifications that consist of error corrections or user features will be implemented under QA processes, but will not be submitted for NRC review.

Duke Code Modification #1 — Allow Access to the Condensation Heat Transfer Correlations With the Forced Convection Heat Transfer Map

Initialization of the ROTSGs using RETRAN-3D uses the forced convection heat transfer map. This is selected by setting variable IHTMAP on the 01000Y card to a value of zero. This standard forced convection option does not allow access to the condensation heat transfer correlations in RETRAN-3D. A code modification was implemented to allow access to the

condensation heat transfer correlations by setting IHTMAP to a value of 2. This option gives the forced heat transfer map, but allows condensation heat transfer to be modeled when appropriate for the local conditions present. This can be important in several transient conditions, such as when the primary water flowing through the steam generator tubes cools to below the secondary saturation temperature. In this situation an appropriate condensation heat transfer coefficient will be selected from the heat transfer correlation set. The technical justification for this code modification is that it allows a correct heat transfer correlation to be used for situations when condensation heat transfer occurs. This modification is functionally equivalent to a similar update made to RETRAN-02, which has been reviewed and approved for use by the NRC.

<u>Duke Code Modification #2</u> Allow the User to Specify the Dittus-Boelter Heat Transfer Correlation for a Specific Conductor

The RETRAN-3D heat transfer correlation package selects an appropriate heat transfer correlation for each conductor surface based on fluid conditions in adjacent volumes. Under some conditions it is useful to be able to select a specific correlation for a given conductor surface rather than using the code-selected correlation. This code modification allows the user to specify either the Dittus-Boelter liquid or vapor correlation for the left surface of a particular conductor. This will then override the code-selected correlation. Word IMCL on Card 15XXXY is set to a value of 41 to specify the liquid correlation, and to a value of 48 to specify the vapor correlation at the left surface of a conductor. The need for this modeling capability arose during analyses of the ROTSG upper shell heat transfer following a steam line break. Due to water carryout into the steam outlet annulus (the volume adjacent to the upper shell), the heat transfer was potentially too high for an analysis in which less heat transfer was conservative. This code modification allowed specifying use of the Dittus-Boelter correlation for vapor to conservatively model the heat transfer to the shell conductor. The code modification also allows specification of the Dittus-Boelter correlation for liquid as another modeling option. The technical justification for this code modification is that a capability to specify a heat transfer correlation for specific applications is appropriate.

<u>Duke Code Modification #3</u>

Duke Code Modification #4

Duke Code Modification #5

Evaluation of RETRAN-3D SER Conditions and Limitations

1. Multidimensional neutronic space-time effects cannot be simulated as the maximum number of dimensions is one. Conservative usage has to be demonstrated.

Staff Position: RETRAN-3D has been modified to include a 3-dimensional nodal kinetics model based on the analytic nodalization method similar to accepted codes. The code has been assessed by calculation of the response of the SPERT prompt-critical tests and has been confirmed by the staff by comparisons with calculations performed with the NESTLE and TORT codes. The staff concludes that the code can adequately predict the response to prompt-critical events such as the PWR rod ejection accident and the BWR rod drop accident. If void generation occurs from an initially un-voided case, the user will have to justify crediting this negative feedback in the analysis.

The code was used by a participant in the Nuclear Energy Agency's International Standard Problem calculation of a hypothetical main steam line break (MSLB) at the Three Mile Island Unit 1 plant. The results of the calculation comparison indicates that RETRAN-3D is comparable to any of the other participating codes.

RETRAN-3D is approved for main steam line break analyses subject to the following conditions. Thermal-hydraulic effects can have a large impact on the cross section evaluation and thus on the resulting power distribution and magnitude. Therefore, the licensee must justify the primary side nodalization for mixing in the vessel and core. The licensee must also evaluate the uncertainties in the modeling.

Duke Position: The RETRAN-3D three-dimensional kinetics model is not used.

2. There is no source term in the neutronics and the maximum number of energy groups is two. The space-time options assume an initially critical system. Initial conditions with zero fission power cannot be simulated by the kinetics. The neutronic models should not be started from subcritical or with zero fission power without further justification.

Staff Position: The basic models in RETPAN-3D are unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not analyze from subcritical or zero fission power initial conditions.

3. A boron transport model is unavailable. User input models will have to be reviewed on an individual basis.

Staff Position: As noted previously in this report, boron transport is handled as a "contaminant" by the "general transport model." This model uses first order accurate upwind difference scheme with an implicit temporal differencing. This approach is well known for being highly diffusive, especially if the Courant limit is exceeded. Since RETRAN-3D has the same model as RETRAN-02 MODOO3 and subsequent versions that have been approved for use, the RETRAN-3D model is also approved with the caveat that the potential to produce misleading results with this scheme necessitates careful review of the results for any case where boron transport/dilution is important.

Duke Position: Previously approved in RETRAN-02 and for Duke applications using RETRAN-02.

4. Moving control rod banks are assumed to travel together The BWR plant qualification work shows that this is an acceptable approximation.

Staff Position: The control bank limitation is applied only to the one-dimensional kinetics model. The staff agrees that the 3-dimensional kinetics model need not be restricted in this way.

Duke Position: Resolved per the Staff Position

5. The metal-water heat generation model is for slab geometry The reaction rate is therefore under-predicted for cylindrical cladding. Justification will have to be provided for specific analyses.

Staff Position: The basic models in RETRAN-3D are unchanged and, therefore, this condition of use applies. However, since RETRAN-3D is not being reviewed for loss-of-coolant accident analysis, where core uncovery and heatup are significant, this condition does not occur in the transients for which application of RETRAN-3D has been reviewed.

Duke Position: Duke does not use the metal-water heat generation model.

6. Equilibrium thermodynamics is assumed for the thermal-hydraulics field equations although there are nonequilibrium models for the pressurizer and the subcooled boiling region.

Staff Position: The RETRAN-3D five equation model permits thermal-hydraulic nonequilibrium between the liquid and vapor phases. While it allows subcooled liquid and saturated steam to be concurrently present, it does not account for subcooled liquid and superheated vapor being concurrently present. Use of the code is not approved for LOCA. Also, the user must be aware of this limitation and avoid conditions which will place subcooled liquid and superheated vapor in contact.

Duke Position: Duke does not use the RETRAN-3D five equation model.

7. While the vector momentum model allows the simulation of some vector momentum flux effects in complex geometry the thermal-hydraulics are basically one-dimensional.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this comment still applies.

Duke Position: As described in Appendix B, Duke is proposing to use the vector momentum option with junction angle input for certain junctions where the momentum flux terms are considered to be potentially important. It is acknowledged that the thermal-hydraulics are basically one-dimensional. Duke's use of this model is based on vendor recommendations.

8. Further justification is required for the use of the homogeneous slip options with BWRs.

Staff Position: RETRAN-3D has five slip equation options for the user to choose from, three of which are retained from RETRAN-02 for compatibility. The recommended model options are based on the Chexal-Lellouche drift flux correlation. The first is the algebraic slip model, which

is approved for use with BWR bundle geometry as given in condition (9). The second is a form of the dynamic slip model that uses the Chexal-Lellouche drift flux correlation to evaluate the interfacial friction approved in condition (10). The user must justify the use of any other slip options.

Duke Position: Duke is not modeling BWRs.

9. The drift flux correlation used was originally calibrated to BWR situations and the qualification work for both this option and for the dynamic slip option only cover BWRs. The drift flux option can be approved for BWR bundle geometry if the conditions of (16) are met.

Staff Position: The Chexal-Lellouche drift flux model has been used in comparisons with FRIGG-2 and FRIGG-4 void fraction data and is acceptable for use in BWR bundle geometry.

Duke Position: Duke is not modeling BWRs.

10. The profile effect on the interphase drag (among all the profile effects) is neglected in the dynamic slip option. Form loss is also neglected for the slip velocity. For the acceptability of these approximations refer to (17).

Staff Position: Form loss terms have been included in the RETRAN-3D dynamic slip model. The Taugl form of the dynamic slip equation also includes profile effects in the interphase drag model. These RETRAN-3D model improvements adequately address the concerns and the model is approved for use when the Chexal-Lellouche model is used to compute the interphase friction. Approval is subject to the conditions given in (16) for the Chexal-Lellouche drift flux correlation. Users must justify use of any other dynamic slip option.

Duke Position: Duke is not using the dynamic slip option.

11. Only one-dimensional heat conduction is modeled. The use of the optional gap linear thermal expansion model requires further justification.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use still applies.

Duke Position: Duke is not using the optional linear gap thermal expansion model.

12. Air is assumed to be an ideal gas with a constant specific heat representative of that at containment conditions. It is restricted to separated and single-phase vapor volumes. There are no other noncondensables.

Staff Position: RETRAN-3D has been extended to include a general noncondensable gas capability which resolves the original concern. However, the noncondensable gas flow model is approved for use subject to the following restriction.

As noted in Section 111.3.0 of the RETRAN-3D Theory Manual (Reference 4), none of the models available for calculating critical flow are appropriate when noncondensable gases are present. Consequently, the code automatically bypasses the critical flow model when noncondensable gases are present in a junction. Users must confirm that noncondensable flows do not exceed appropriate critical flow values or justify use of values that may exceed critical flow values.

Duke Position: Duke is not using the noncondensable gas flow model.

13. The use of the water properties polynomials should be restricted to the subcritical region. Further justification is required for other regions.

Staff Position: For enthalpies less than approximately 820 Btu/lbm, the difference between the ASME and RETRAN-3D curve fit values of the specific volume range from less than 0.2 percent to approximately 1.3 percent for pressures ranging from 0.1 to 6,000 psia. Further, for enthalpies greater than 820 Btu/lbm and pressures greater than 4200 psia, the differences in specific volume are also less than 1.0 percent. RETRAN-3D is approved for use with PWR ATWS analyses where the peak pressure resides in the regions described above.

For enthalpies greater than 820 Btu/lbm and pressures between 3200 and 4200 psia, the differences in specific volume increase as the enthalpy increases and the pressure decreases. The maximum error of approximately 3.8 percent occurs at the critical point. PWR ATWS analysis using RETRAN-3D in this region will require additional justification that the difference in specific volume does not adversely affect the calculation of the peak pressure.

Duke Position: Duke will address the above condition if an application encounters conditions in the region of concern.

14. A number of regime-dependent minimum and maximum heat fluxes are hardwired. The use of the heat transfer correlations should be restricted to situations where the pre-CHF heat transfer or single-phase heat transfer dominates.

Staff Position: RETRAN-3D contains both the "forced convection option" contained in RETRAN-02 which is the basis for this restriction, and a second option referred to as the "combination heat transfer map." If the first option is chosen, the "forced convection option," approval is granted only for use in pre-CHF and single-phase heat transfer regimes. If the second option is chosen, the "combination heat transfer map," then there are no discontinuities between successive heat transfer regimes and the appropriate heat transfer value should result. Therefore, the combination heat transfer option is approved for use.

Duke Position: Section 2.2.7.5 of Duke Power topical report DPC-NE-3000-PA, "Thermal-Hydraulic Transient Analysis Methodology," [

] Note that Duke has implemented, and the NRC has approved, a RETRAN-02 code modification to enable accessing the condensation heat transfer correlations with the forced convection heat transfer map. A similar code modification has been implemented for RETRAN-3D.

15. The Bennett flow map should only be used for vertical flow within the conditions of the data base and the Beattie two-phase multiplier option requires qualification work

Staff Position: The Beattie two-phase multiplier has been removed from RETRAN-3D. The Govier horizontal flow map has been added to supplement the Bennett map for vertical flow and is acceptable.

Duke Position: Resolved per the Staff Position.

16. No separate effects comparison have been presented for the algebraic slip option and it would be prudent to request comparisons with the FRIGG tests before the approval of the algebraic slip option.

Staff Position: The algebraic slip option has been modified to include the Chexal-Lellouche drift flux model. Use of the Chexal-Lellouche drift flux model for BWR and PWR applications within the range of conditions covered by the steam-water database used to develop and validate the model is approved. The model has been qualified with data from a number of steady-state and two-component tests. While the small dimensions of the fuel assembly are covered, as noted previously in this safety evaluation, the data for large pipe diameters, such as reactor coolant system pipes, are not extensive and use of the Chexal-Lellouche model will need justification. Assessment work indicates that the model tends to under-predict the void profile in the range of 12 to 17 MPa. In addition, the accuracy of the model in the range of 7.5 to 10 Mpa, which covers BWR ATWS conditions, has not been fully demonstrated. Results of analyses using the model in these ranges must be carefully reviewed.

The Chexal-Lellouche correlation cannot be used in situations where CCFL is important unless validation for appropriate geometry and expected flow conditions is provided.

Duke Position: [

] This use is consistent with the Staff Position.

17. While FRIGG test comparisons have been presented for the dynamic slip option the issues concerning the Schrock-Grossman round tube data comparisons should be resolved before the dynamic slip option is approved. Plant comparisons using the option should also be required.

Staff Position: Assessment analyses (Reference 4), have shown that "the issues concerning the Schrock-Grossman round tube data comparisons" (actually the Bennett round tube data) are due to early prediction of CHF, which is nearly independent of the slip model used. Since the issue raised in the limitation is not related to the dynamic slip model, the limitation is considered to be resolved. The dynamic slip model is approved for use as given in condition (10).

Duke Position: Resolved per the Staff Position.

18. The nonequilibrium pressurizer model has no fluid boundary heat losses, cannot treat thermal stratification in the liquid region and assumes instantaneous spray effectiveness and a constant rainout velocity. A constant UA is used and flow detail within the component cannot be simulated. There will be a numerical drift in energy due to the inconsistency between the two-region and the mixture energy equations but it should be small. No comparisons were presented involving a full or empty pressurizer. Specific application of this model should justify the lack of fluid boundary heat transfer on a conservative basis.

Staff Position: The concern raised in this limitation of use is partially resolved in RETRAN-3D. Wall heat transfer can be included in the RETRAN-3D pressurizer model. Including wall heat transfer resolves this concern.

While the model does not directly account for thermal stratification, its effects can be included by use of normal nodes below the pressurizer volume. The user will have to justify the lack of thermal stratification or the use of normal nodes below the pressurizer should there be an indication that it would be important in the analysis.

The mixture and two-region energy equations are consistent for the implicit solution method where the mixture energy equation is used with the vapor-region energy equation. This eliminates inconsistency between the two-region and mixture energy equations and the concern regarding a potential drift in the region energies.

The staff notes that when a pressurizer fills or drains, a single region exists for which the normal pressure equation of state is used. Lack of numerical discontinuities in validation analyses of filling and draining pressurizers indicates that the model is functioning properly. It is the responsibility of the code user to justify any numerical discontinuity in the pressurizer during a filling or draining event.

The pressurizer model has options that require user-supplied parameters. Users must provide justification for these model parameters.

Duke Position: Duke is not proposing any changes in modeling the pressurizer with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA. The good modeling practices in the Staff Position are noted.

19. The non-mechanistic separator model assumes quasi-statics (time constant approximately few tenths of seconds) and uses GE BWR6 carryover/carryunder curves for default values. Use of default curves has to be justified for specific applications. As with the pressurizer a constant L/A is used. The treatment in the off normal flow quadrant is limited and those quadrants should be avoided. Attenuation of pressure waves at low flow/low quality conditions are not simulated well. Specific applications to BWR pressurization transients under those conditions should be justified.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not use the separator model.

20. The centrifugal pump head is divided equally between the two junctions of the pump volume. Bingham pump and Westinghouse pump data are used for the default single-phase homologous curves. The SEMISCALE MOD-1 pump and Westinghouse Canada data are for the degradation multiplier approach in the two-phase regime. Use of the default curves has to be justified for specific applications. Pump simulation should be restricted to single-phase conditions.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: As noted in the Staff Position, the basic model in RETRAN-3D is unchanged from RETRAN-02. Duke is not proposing any changes in modeling the reactor coolant pumps with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA, Section 2.2.6.2.

21. The jet pump model should be restricted to the forward flow quadrant as the treatment in the other quadrants is conceptually not well founded. Specific modellng of the pump in terms of volumes and junctions is at the user's discretion and should therefore be reviewed with the specific application.

Staff Position: Subsequent revisions of RETRAN-02 addressed this limitation. Since RETRAN-3D has the same model as RETRAN-02 MODOO3, and subsequent versions, their acceptance applies to RETRAN-3D.

Duke Position: Duke does not model BWR jet pumps.

22. The non-mechanistic turbine model assumes symmetrical reaction staging, maximum stage efficiency at design conditions, a constant UA and a pressure behavior dictated by a constant loss coefficient It should only be used for quasi-static conditions and in the normal operating quadrant.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not use the turbine model.

23. The subcooled void model is a nonmechanistic profile fit using a modification of EPRI recommendations for the bubble departure point It is used only for the void reactivity computation and has no direct effect on the thermal-hydraulics. Comparisons have only been presented for BWR situations. The model should be restricted to the conditions of the qualification data base. Sensitivity studies should be requested for specific applications. The profile blending algorithm used will be reviewed when submitted as part of the new manual (MODOO3) modifications.

Staff Position: The profile blending algorithm approved for RETRAN-02 MODOO3 is used in RETRAN-3D therefore this condition has been satisfied.

Duke Position: Resolved per the Staff Position.

24. The bubble rise model assumes a linear void profile, a constant rise velocity (but adjustable through the control system), a constant UA, thermodynamic equilibrium, and makes no attempt to mitigate layering effects. The bubble mass equation assumes zero junction slip which is contrary to the dynamic and algebraic slip model. The model has limited application and each application must be separately justified.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies. However, the layering effects encountered in RETRAN-02 can be eliminated using the RETRAN-3D stack model. This partially resolves the concern by resolving the layering limitation through use of the stack model.

Duke Position: As noted in the Staff Position, the basic model in RETRAN-3D is unchanged from RETRAN-02. Duke is not proposing any changes in using the bubble rise model with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA, Section 2.2.6.4. Duke does not currently stack bubble rise volumes, but if future modeling does, the stack model will be used.

25. The transport delay model should be restricted to situations with a dominant flow direction.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies. The appropriate application of the model is for one-dimensional flow. The user will have to justify use of this option in the absence of a dominant flow direction.

Duke Position: As noted in the Staff Position, the basic model in RETRAN-3D is unchanged from RETRAN-02. Duke is not proposing any changes in using the transport delay model with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA, Section 2.2.7.4. The limitation with applying this model without a dominant flow direction is well known and is avoided.

26. The stand-alone auxiliary DNBR model is very approximate and is limited to solving a one-dimensional steady-state simplified HEM energy equation. It should be restricted to indicating trends.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not use the auxiliary DNBR model.

27. Phase separation and heat addition cannot be treated simultaneously in the enthalpy transport model For heat addition with multidirectional, multifunction volumes the enthalpy transport model should not be used without further justification. Approval of this model will require submittal of the new manual (MODOO3) modifications.

Staff Position: A number of the simplifying assumptions in the RETRAN-02 enthalpy transport model have been eliminated in RETRAN-3D which now allows multiple inlet and outlet flows

and eliminates the simplifying assumptions related to mass distribution and pressure change effects. This condition has been adequately addressed.

Duke Position: Resolved per the Staff Position

28. The local conditions heat transfer model assumes saturated fluid conditions, one-dimensional heat conduction and a linear void profile. If the heat transfer is from a local condition volume to another fluid volume, that fluid volume should be restricted to a non-separated volume. There is no qualification work for this model and its use will therefore require further justification.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position:

29. The initializer does not absolutely eliminate all ill-posed data and could have differences with the algorithm used for transient calculations. A null transient computation is recommended. A heat transfer surface area adjustment is made and biases are added to feedwater inlet enthalpies in order to justify steady-state heat balances. These adjustments should be reviewed on a specific application basis.

Staff Position: The over-specified condition is identified by the RETRAN-3D steady-state input checking, resolving the concern regarding ill-posed data. The user must still run null transients to ensure that unwanted control or trip actions are not affecting the transient solution.

RETRAN-3D has available a low power steady-state steam generator initialization option that eliminates the heat conductor area change used in the RETRAN-02 initialization scheme. When this option is used, no adjustments are made to the heat transfer area and this specific concern is resolved. However, either the pressure or temperature is adjusted on the secondary side. These adjustments should be reviewed by the user on a specific application basis. The low power steady-state initialization option is approved for use.

Duke Position: Resolved per the Staff Position.

30. Justification of the extrapolation of FRIGG data or other data to secondary-side conditions for PWRs should be provided. Transient analysis of the secondary side must be substantiated. For any transients in which two-phase flow is encountered in the primary all the two-phase flow models must be justified.

Staff Position: The Chexal-Lellouche correlation is approved for use with PWR applications as stated in conditions (10) and (16). The user must justify choosing any other two-phase flow correlation.

Duke Position: [.]
Resolved per the Staff Position.

31. The pressurizer model requires model qualification work for the situations where the pressurizer either goes solid or completely empties.

Staff Position: The pressurizer model is approved for use with filling and draining events as given in condition (18).

Duke Position: Resolved per the Staff Position

32. Transients which involve three-dimensional space-time effects such as rod ejection transients would have to be justified on a conservative basis.

Staff Position: The 3-dimensional kinetics model, as noted in limitation 1 above, satisfies this limitation.

Duke Position: Resolved per the Staff Position. Duke is not using the three-dimensional model.

33. Transients from subcritical, such as those associated with reactivity anomalies should not be run.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke is not running any transients from subcritical.

34. Transients where boron injection is important, such as steamline break will require separate justification for the user-specified boron transport model.

Staff Position: The generalized transport model was added to RETRAN-3D to provide the capability to track materials such as boron. Specific application of the model to steam line break transients must be justified by the user. The model is approved for use as given in condition (3).

Duke Position: The generalized transport model is unchanged from RETRAN-02 relative to its use for modeling boron transport. The generalized transport model was approved in the SER for RETRAN-02 MOD005.0 dated November 1, 1991. Duke's use of this model for Oconee

emergency boron injection is documented in topical report DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology," Revision 1, Section 15.3.1.1.3. This topical report was approved by the NRC with SERs dated October 1, 1998 and May 25, 1999. No further NRC review is necessary.

35. For transients where mixing and cross flow are important, the use of various cross flow loss coefficients has to be justified on a conservative basis.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke models mixing and cross flow in the reactor vessel during transients and accidents in which loop asymmetry is important. Duke's modeling for Oconee is documented in topical report DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology," Revision 1, Section 15.2.1.1. This topical report was approved by the NRC with SERs dated October 1, 1998 and May 25, 1999. No further NRC review is necessary.

36. ATWS events will require additional submittals.

Staff Position: RETRAN-3D is approved for PWR ATWS analyses as given in condition (13).

Duke Position: Resolved per the Staff Position.

37. For PWR transients where the pressurizer goes solid or completely drains, the pressurizer behavior will require comparison against real plant or appropriate experimental data.

Staff Position: The pressurizer model is approved for use with filling and draining events as noted in the discussion of conditions (18) and (31).

Duke Position: Resolved per the Staff Position

38. PWR transients, such as steam generator tube rupture, should not be analyzed for two-phase conditions beyond the point where significant voiding occurs on the primary side.

Staff Position: The use of slip models for PWR applications is approved for use as given in conditions (16) and (30).

Duke Position: In the Oconee steam generator tube rupture UFSAR Chapter 15 analysis significant voiding does not occur on the primary side. However, significant voiding can occur on the primary side for steam line break events. Duke's UFSAR Chapter 15 steam line break modeling for Oconee is documented in topical report DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology," Revision 1, Chapter 15. This topical report was approved by the NRC with SERs dated October 1, 1998 and May 25, 1999. Duke's UFSAR Chapter 6 steam line break mass and energy release modeling for Oconee is documented in topical report DPC-NE-3003-PA, "Mass and Energy Release and Containment Response Methodology," Chapter 5. This topical report was approved by the NRC with SER dated March 15, 1995. Modeling of two-

[

] No further NRC review is necessary.

39. BWR transients were asymmetry leads to reverse jet pump flow such as the one recirculation pump trip, should be avoided.

Staff Position: As noted in the discussion of condition (21), this is resolved.

Duke Position: Duke does not model BWRs.

40. Organizations with NRC-approved RETRAN-02 methodologies can use the RETRAN-3D code in the RETRAN-02 mode without additional NRC approval, provided that none of the new RETRAN-3D models listed in the definition are used. Organizations with NRC-approved RETRAN-02 methodologies must obtain NRC approval prior to applying any of the new RETRAN-3D models listed above for UFSAR Chapter 15 licensing basis applications. Organizations without NRC-approved RETRAN-02 methodologies must obtain NRC approval for such methodologies or a specific application before applying the RETRAN-02 code or the RETRAN-3D code for UFSAR Chapter 15 licensing basis applications. Generic Letter 83-11 provides additional guidance in this area. Licensees who specifically reference RETRAN-02 in their technical specifications will have to request a Technical Specification change to use RETRAN-3D.

Duke Position: The submittal of DPC-NE-3000-P, Revision 3, includes the use of

The SER on p. 33 also states that use of the "new control blocks added to improve functionality" requires NRC review and approval. The new control blocks in RETRAN-3D are the following:

ABS - Absolute value
F2D - Two-dimensional interpolation
RAT - Rate
STF - Second-order transfer function

None of these new control block models have yet been incorporated into any of the Duke RETRAN models used for licensing basis applications. However, use of some of these new control block models in the future is likely to enhance and simplify applications. Since all of these new control blocks consist of well-founded arithmetic and mathematical formulas, similar to the control blocks included in RETRAN-02, it is not understood why NRC approval prior to their use is necessary. Duke requests NRC approval to use the new RETRAN-3D control blocks for future applications consistent with their formulation.

41. RETRAN may be used for BWR ATWS subject to the following restrictions: The licensee must validate the chosen void model over the range of pressure, channel inlet flow, and inlet subcooling encountered during the transient that are outside the range of conditions for which assessment is available. Furthermore, the licensee should validate the choice of steam separator model and evaluate its use relative to steam separator performance data relevant to the conditions

present during the ATWS simulation. The licensee must also evaluate the uncertainties in the modeling. See Condition (16) and the Staff Position for related information.

Duke Position: Duke does not model BWRs.

42. The RETRAN-3D five-equation, or nonequilibrium, model uses flow regime maps and flow pattern dependent heat transfer and interfacial area models to simulate the heat and mass transfer processes between phases. A licensee wishing to apply the five-equation model will have to justify its use outside areas of operation where assessment has been documented. This may include either separate effects or integral systems assessment that cover the range of conditions encountered by the application of interest. An assessment of the uncertainties must also be provided. The model is approved subject to these conditions.

Duke Position: Duke does not use the five-equation model for licensing basis applications.

43. Assessment performed in support of use of RETRAN-3D must also address consistency between the RETRAN-3D calculations and any auxiliary calculations that are a part of the overall methodology, such as, departure from nucleate boiling or critical power ratio.

Duke Position: Duke uses results from RETRAN-3D analyses for input to other codes to perform core power distribution analyses, detailed core thermal-hydraulic analysis of the departure from nucleate boiling phenomenon, fuel rod and pellet thermal and mechanical behavior analyses, and containment thermal and structural response to high-energy line breaks. The details of these other methodologies have been submitted and approved by the NRC as appropriate. Any revisions to these methodologies, including any changes due to the use of RETRAN-3D in place of RETRAN-02, will be submitted for NRC review prior to their use for licensing basis applications.

44. The staff concludes that the lack of a detailed RETRAN-3D specific user guideline document mandates a statement on the user's experience and qualification with the code when analyses are submitted in support of licensing actions. This statement is expected to be consistent with the guidance of Generic Letter 83-11.

Duke Position: It is noted that Volume 3 of the EPRI RETRAN-3D documentation set has been enhanced subsequent to the NRC SER to include a significant amount of user guidelines regarding modeling option selection, in particular for the new RETRAN-3D models and options. Revision 3 to DPC-NE-3000-P fully describes Duke's use of the RETRAN-3D code for simulating the Oconee Nuclear Station with replacement steam generators. This revision is submitted for NRC review with the intent of maintaining the documentation of the Duke RETRAN methodology current, along with the main purpose of obtaining NRC review and approval for the transition from RETRAN-02 to RETRAN-3D for Oconee. This topical report revision extends Duke's response to Generic Letter 83-11. Duke's current level of RETRAN user experience is 15 engineers with a total of 144 years of experience with RETRAN-02 and RETRAN-03/-3D.

45. Assessment of the RETRAN-3D code for the models not explicitly approved in this safety evaluation will be the responsibility of the licensee or applicant. In addition, application of the

RETRAN-02 or RETRAN-3D codes for best estimate analysis of UFSAR Chapter 15 licensing basis events may require additional code and model assessment, and an evaluation of uncertainties to assure accurate prediction of best estimate response. This condition is based on the absence, in the best estimate analysis approach, of the conservative assumptions in traditional UFSAR Chapter 15 licensing basis analyses. For each use of RETRAN-3D in a licensing calculation, it will be necessary for a valid approach to assessment to be submitted, which is expected to include a PIRT for each use of the code and the appropriate assessment cases and their results. The scope of the PIRT and validation/assessment will be commensurate with the complexity of the application.

<u>Duke Position</u>: Duke has previously received NRC review and approval for application of the RETRAN-02 code to the licensing basis applications for non-LOCA transients and accidents for the Oconee Nuclear Station. The three RETRAN-related topical reports and associated NRC SERs supporting Oconee are:

> DPC-NE-3000-PA, Revision 2, "Thermal-Hydraulic Transient Analysis Methodology", December 2000. SERs are dated 11/15/91 (Revision 0), 8/8/94 (Revision 1), and 12/27/95 (Revision 2)
>
> DPC-NE-3003-PA, "Mass and Energy Release and Containment Response Methodology", November 1997. SER is dated 3/15/95
>
> DPC-NE-3005-PA, Revision 1, "UFSAR Chapter 15 Transient Analysis Methodology, August 1999. SERs are dated 10/1/98 (Revision 0) and 5/25/99 (Revision 1)

These topical reports have all being revised and submitted for NRC review to address the Oconee replacement steam generators and use of the RETRAN-3D code for Oconee non-LOCA transient and accident analyses. Based on the close similarity of the replacement and original steam generators, the transient thermal-hydraulic behavior will be very similar. The only significant difference will be for the main steam line break analysis, in which the flow restricting orifices in the replacement steam generators steam outlet nozzles will effectively reduce the maximum break size and the blowdown rate.

[ ] That assessment

activity was presented and reviewed by the NRC during the review of earlier revisions of DPC-NE-3000. [ ] has been incorporated into the Duke version of the RETRAN-3D code as described in Revision 3 to DPC-NE-3000-P.

In summary, Duke has previously obtained NRC review for RETRAN-02 modeling of Oconee with the original steam generators. Substantial validation and assessments comparisons were associated with the previous revisions to DPC-NE-3000. The designs of the original and replacement steam generators are very similar, and the transient performance will be very similar except for the response to large steam line break accidents. Revision 3 describes the use of RETRAN-3D for modeling Oconee with replacement steam generators. [ ] Duke is not proposing to use this model for best-estimate licensing applications. The traditional conservative approach will continue to be used for licensing applications. A PIRT is not being submitted due to the previous NRC approval of the Duke RETRAN methodology topical reports, the limited scope of changes in the methodologies, the similarity of the designs of the new and replacement Oconee steam generators, the use of only one new RETRAN-3D model, and the assessment that has been performed to justify use of the one new model.

References

B-1 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

B-2 RETRAN-3D MOD003.1DKE, SDQA-30218-NGO, Duke Power, October 30, 2001

APPENDIX C

GOTHIC VERSION 7.0 CODE

C.1 Code Description

The GOTHIC Version 7.0 code (Reference C-1), developed by Numerical Applications, Inc. under contract to the Electric Power Research Institute (EPRI), is used for performing thermal-hydraulic analysis of nuclear power plant containment and auxiliary buildings. The GOTHIC 7.0 code was developed from the FATHOMS code (Reference C-2), which in turn was developed from the COBRA-NC code (Reference C-3). Due to the close similarities between GOTHIC 7.0 and FATHOMS, the content of this Appendix is very similar to Chapter 2.4. GOTHIC 7.0 solves the conservation equations for mass, energy, and momentum for multi-component, two-phase flow. These conservation equations are solved numerically on a finite-volume mesh made up of numerous computational cells. The code features a nodalization scheme in which lumped parameter, one-, two-, or three-dimensional analysis or any combination of these may be performed. Velocity fields are included for three phases:

- Vapor / non-condensable gas mixture
- Continuous liquid
- Liquid droplet

Interfacial heat, mass, and momentum transfer are modeled, and the effects of two-phase slip on pressure drop are also accounted for. Three temperature fields are included:

- Vapor / non-condensable gas mixture
- Continuous liquid
- Liquid droplet

These fields may be in thermal non-equilibrium within the same calculational volume. A total of eight non-condensable gases may be modeled, with mass transport equations solved for each component of the non-condensable gas mixture.

GOTHIC 7.0 includes finite-difference conduction models for passive thermal conductors. Concrete walls and structural steel within the Reactor Building are simulated with these models. These conductors may have the following geometrical shapes:

- Flat plate (wall)
- Cylindrical tube
- Solid rod

Heat is conducted through the thickness or radius of each of these conductor models. They may be thermally connected to fluid volumes on either surface. A variety of condensation heat transfer coefficient options are available for the surfaces of these thermal structures. Heaters and coolers may also be modeled within calculational volumes in GOTHIC 7.0. Any volumetric heat sources or cooling equipment may be simulated with these models.

Each GOTHIC 7.0 analysis model consists of a network of volumes connected by flow paths. Typically, each volume represents a room or area within the Reactor Building. Any volume may be partitioned into a two- or three-dimensional mesh for which mass, energy, and momentum balance are maintained. This allows the fluid property distribution within a room to be calculated and flow patterns within a room to be predicted. The flow paths represent doorways, pipes or piping systems, vents, etc. Within each subdivided volume, the momentum equations for the vapor/gas mixture, droplets, and liquid are solved for all flow paths, with each phase having its own velocity. A variety of mechanical components to simulate various plant equipment can be modeled along these flow paths. These include:

- Valves
- Pumps
- Heat exchangers
- Spray nozzles
- Volumetric fans

Boundary conditions for the containment model are implemented using these flow paths. The user may specify mass and energy sources or sinks with these boundary conditions. The boundary conditions may be coupled in groups to model flowstreams which are sent to different calculational volumes. The user has the ability to model cooling and spray systems associated with containment safety features using these options.

GOTHIC 7.0 features the use of control variables that are not available in FATHOMS. The control variables provide enhanced capability and flexibility to input boundary conditions or forcing functions into the code. For instance, the output of the BFLOW code to determine the quality of water exiting a cold leg break for long-term mass and energy releases can be input via control variables. This capability was not available for FATHOMS and specific code changes were implemented in order to input the BFLOW code results (Section 2.4.1.2).

The GOTHIC 7.0 code was released by EPRI in 2001 to provide EPRI members with an efficient, state-of-the-art containment analysis code. The code had undergone extensive testing and benchmarking by Numerical Applications, Inc. prior to release.

C.2 Simulation Model

This section gives a brief description of the simulation models used for the GOTHIC 7.0 containment analyses. The nodalization diagram is the same as for the FATHOMS base model shown in Figure 2.4-1.

Nodalization

Large dry containments have traditionally been modeled with one large volume, with all pressures and temperatures being considered volume-averaged quantities. The Oconee GOTHIC 7.0 base model uses a similar modeling approach. As in the FATHOMS model, a separate sump volume is represented in the GOTHIC 7.0 base model. This consists of the lowest 8.6 feet of the containment space, and the upper 195 feet representing the remainder of the containment space. These two nodes are referred to here as the atmosphere region and the sump region. The sump temperature refers to the liquid temperature in the sump volume, and the vapor temperature refers to the temperature of the vapor phase of the atmosphere volume. There is a small vapor space in the sump region, but it is so small that it does not affect the results of applications of the model.

The containment free volume and passive heat sink data used for the base model include a 1% conservative allowance for uncertainty.

Initial Conditions

The initial pressure and temperature assumed in the Reactor Building varies with each analysis, depending on the purpose of the analysis. For the peak pressure analysis, it was determined that variations in the initial air mass had a large impact on the calculated peak pressure. A higher initial air mass leads to higher calculated peak pressures. The initial air mass can be increased by assuming a low initial building temperature The early time at which the peak pressure occurs (within 30 seconds for each limiting large break LOCA case) means that the increased initial air mass had a larger effect than the decreased heat transfer into the passive heat structures at a higher initial temperature. Therefore, conservatively low initial temperatures are assumed in the peak pressure analyses. In all long-term analyses, as well as the steam line break analysis, a high initial building temperature is assumed.

Likewise, the initial Reactor Building humidity is assumed to be at 0% to maximize the air mass in the peak pressure analyses, and at 100% in the long-term analyses. The initial building humidity has very little impact on the results of either type of analysis.

A high initial Reactor Building pressure is conservatively assumed for all peak pressure analyses. This maximizes the calculated peak building pressure. In the long-term analyses, this initial pressure has no significant impact on results.

Heat Structures

The concrete and steel heat structures present in the Oconee containment building are modeled using the heat slab models in GOTHIC 7.0. They are modeled with an insulated boundary condition for one surface. For the building walls, dome, and base, it would be many days before significant quantities of heat would conduct through the thickness of the structure, so this insulated boundary condition is used on each outer surface. The remaining structures, all internal to the Reactor Building, are exposed to the containment atmosphere on all sides, but the entire surface area is combined onto one surface for the GOTHIC 7.0 heat slab model. The zero heat fluxes on the second surface here represent a symmetry condition at the midplane of the structure. The slabs are modeled in full thickness, however, so that the total volume will be conserved. All structures are initially at the initial Reactor Building temperature.

All structures are modeled as walls, with one-dimensional heat transfer occurring in the direction perpendicular to the structure surface. The noding divisions are finer closer to the surface, where larger temperature gradients are expected.

The Uchida heat transfer correlation is used for heat transfer between the structure subcooled surfaces and the containment atmosphere. The standard heat transfer logic present in GOTHIC 7.0 is used for other heat transfer modeling. For the Reactor Building floor, which would be covered with a pool of water, a constant heat transfer coefficient of 20 Btu/hr-ft^2-^{o}F is assumed.

Reactor Building Cooling Units

The Reactor Building Cooling Units (RBCUs) are safety-grade fan coolers modeled using the standard cooler model in GOTHIC 7.0. Input parameters are chosen from design values for the coolers. The heat removal characteristics of the coolers varies with each analysis. The coolers are typically actuated at 9 psig Reactor Building pressure, plus a 300 second delay time. These setpoints include allowances for instrument uncertainties and all other associated delays.

Reactor Building Spray System

The Reactor Building Spray System (RBS) is modeled with a single junction for injection from the borated water storage tank (BWST), or with a set of junctions for sump recirculation mode. The RBS flow rate varies depending on the purpose of the analysis. Uncertainty in the RBS flow rate is included.

All spray is assumed to enter the containment atmosphere with an average droplet size of 700 μm. No spray efficiency parameter is necessary in the GOTHIC 7.0 flow models. The spray is typically actuated at 30 psig Reactor Building pressure, with a delay time of 92 seconds for the spray header fill time. These setpoints include allowances for instrument uncertainties and all other associated delays.

C.3 Code Validation

The GOTHIC 7.0 code authors, Numerical Applications, Inc., subjected the code to extensive benchmarking efforts throughout the process of refining and updating the code. The code has been compared against analytic solutions as well as experimental data (Reference C-1). This set of tests includes comparisons with data from the Battelle Frankfurt HDR containment experimental facility, the Hanford Engineering Development Laboratory (HEDL) facility, and many other tests. The validation effort included a comparison against a number of separate effect test, such as fan coolers, pump head vs. flow tests, jet breakup tests and other specially designed analytic problems to test specific components and GOTHIC 7.0 code capabilities.

Of special importance is the comparison by NAI of GOTHIC 7.0 predictions with the Carolina Virginia Tube Reactor (CVTR) results. This set of data is particularly appropriate for this purpose because it applies to the pressurization of a large, dry containment building similar to that of Oconee. These experiments studied the pressurization of the CVTR building due to a saturated steam blowdown, with passive heat structures and building spray aiding in the depressurization. A GOTHIC 7.0 model was developed for the CVTR containment building, and simulations of three steam blowdown tests were performed. The pressure and temperature responses for all three tests are predicted reasonably well.

The GOTHIC 7.0 code is an extension of the COBRA-NC code and retains all of the modeling capabilities of COBRA-NC. The COBRA-NC code was also benchmarked against analytic solutions and experimental data, including steam and steam/water blowdowns in the Battelle Frankfurt Model Containment, Containment Analysis Standard Problems 1, 2, and 3, blowdown tests in the HDR facility and hydrogen distribution tests in the Battelle Frankfurt Model Containment (Volume 7, Reference C-3). These verification results can be reasonably assumed to valid for the GOTHIC 7.0 code as well. All efforts have validated the code's ability to accurately predict pressure and temperature transients within containment structures.

3.4 Model Validation

After a GOTHIC 7.0 model of the Oconee containment was developed, efforts to compare the results obtained from that model were undertaken. Since no test data exists for the pressurization rate of the Oconee Reactor Building following a high-energy line break, an available means of validation of the GOTHIC 7.0 model for the Oconee design is a code-to-code comparison to the results of a FATHOMS LOCA analysis. The FATHOMS analyses was compared previously (Chapter 2.4) with the CONTEMPT predictions performed by Babcock and Wilcox (B&W) for the original Oconee FSAR. An analysis using mass and energy release data for a 14.1 ft^2 hot leg break, representing a double-ended guillotine break of the hot leg, is used. This is the bounding peak containment pressure transient for Oconee according to the existing FATHOMS licensing basis analyses in the UFSAR.

The Oconee GOTHIC 7.0 model, input data, and code option selection, as well as the boundary conditions, are identical to the corresponding FATHOMS analysis to the extent possible. This approach is intended to allow a code-to-code comparison.

The results of this comparison are shown in Figures C-1 and C-2. The containment pressure curves (Figure C-1) show an identical trend for both codes. The peak pressure calculated by GOTHIC 7.0 is approximately 1 psi lower than the FATHOMS result. The containment temperature profile (Figure C-2) is identical for both codes. These results show that the GOTHIC 7.0 code and the Oconee GOTHIC model provide essentially identical results when compared to the FATHOMS code and model in the DPC-NE-3003 methodology. Use of either code and model in the future is technically equivalent. The intent is to begin a transition from FATHOMS to GOTHIC 7.0 upon approval of DPC-NE-3003-P, Revision 1.

C.5 References

C-1 NAI 8907-02 Rev 13, “GOTHIC Containment Analysis Package – User Manual”, Version 7.0, July 2001

C-2 CAP-Containment Analysis Package (FATHOMS 2.4), Numerical Applications, Inc., October 10, 1989

C-3 NUREG/CR 3262, "COBRA-NC: A Thermal Hydraulics Code for Transient Analysis of Nuclear Reactor Components", Volumes 1 to 7, Wheeler, et.al., Pacific Northwest Laboratory, April 1986.

GOTHIC 7.0 - FATHOMS Comparison For
Oconee Large Break LOCA (14.1 ft^2 break - S/G Inlet)

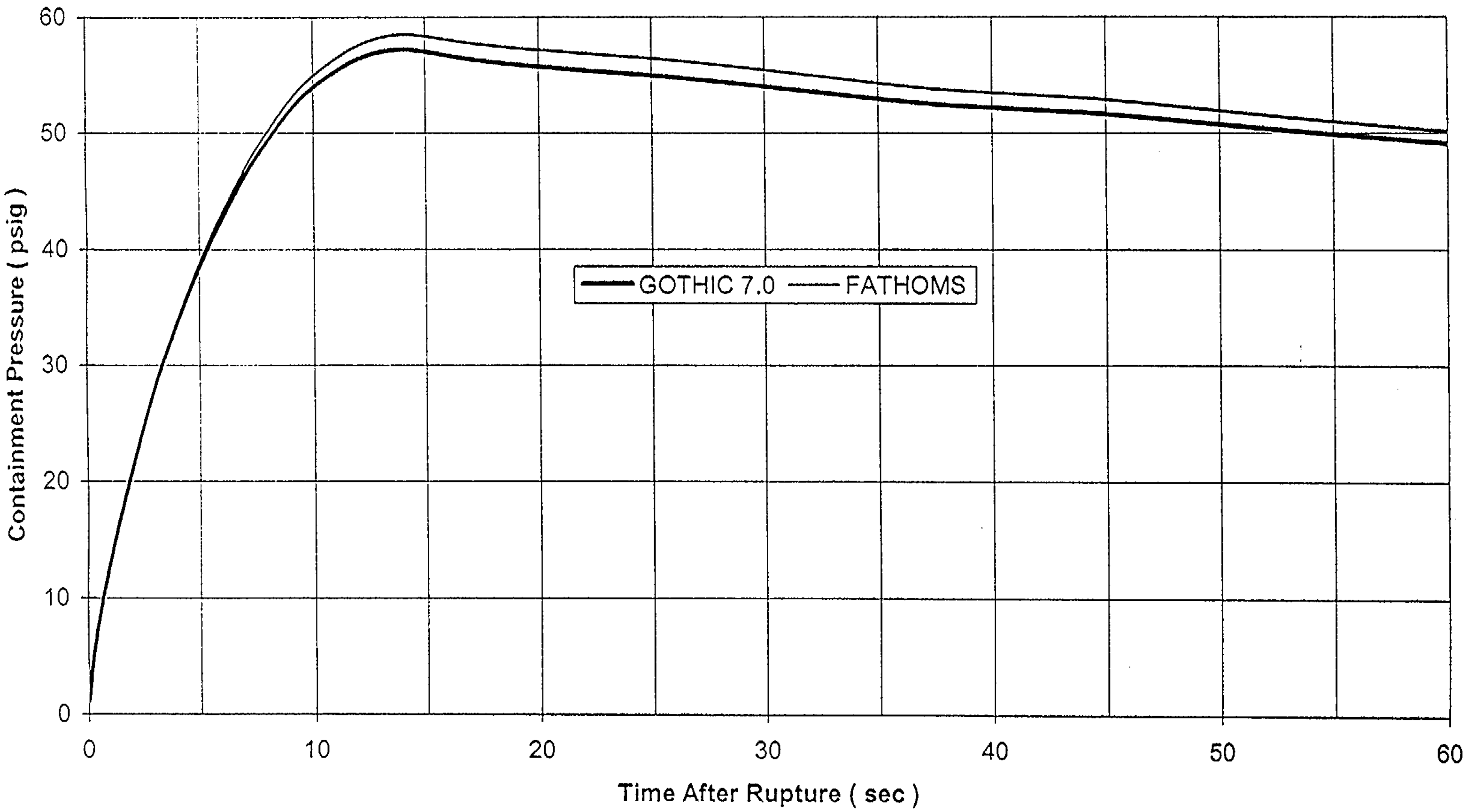


FIGURE C-1

GOTHIC 7.0 - FATHOMS Comparison For
Oconee Large Break LOCA (14.1 ft2 break - S/G Inlet)

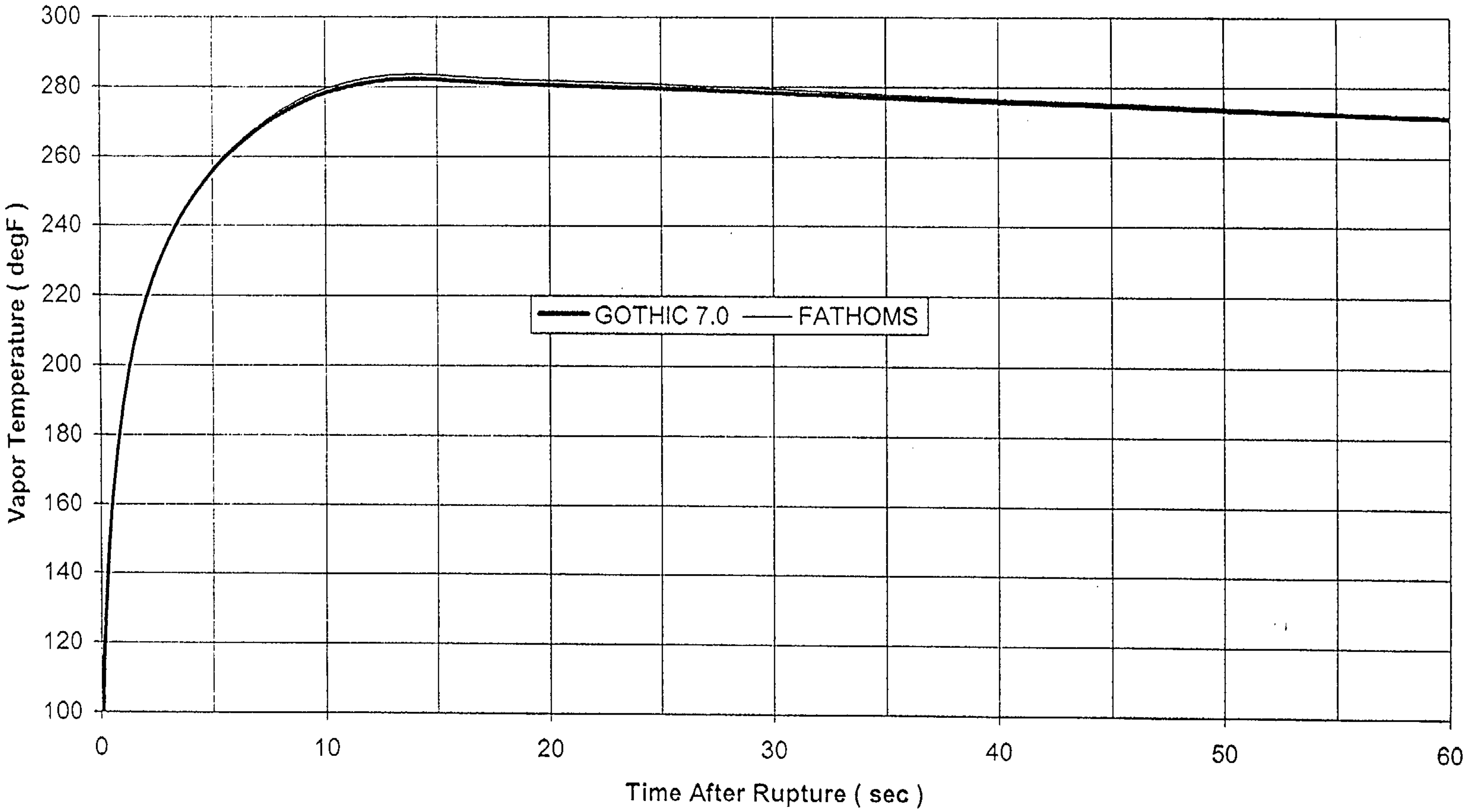


FIGURE C-2

Attachment 6

UFSAR Chapter 15 Transient Analysis Methodology

DPC-NE-3005-P (Non-Proprietary)
Revision 2

June 2002

Nuclear Engineering Division
Nuclear Generation Department
Duke Power Company

DPC-NE-3005 Non-Proprietary, Revision 2

Description and Technical Justification

Revision 2 to DPC-NE-3005 consists of changes necessary to analyze the UFSAR Chapter 15 non-LOCA transients and accidents with RETRAN-3D and with the replacement once-through steam generators (ROTSGs), minor technical and editorial changes to update the report since Revision 1 was submitted in 1999, and a statement of the compliance with the NRC's SER on RETRAN-3D. Each of the revisions is described in detail, and model changes are supported by technical justification. The Oconee RETRAN-3D model (DPC-NE-3000-P, Revision 3, June 2002) has been reviewed by Computer Simulation & Analysis (CSA), Inc., the developer of the RETRAN-3D code. There are currently no other organizations using RETRAN-3D for modeling B&W-designed reactors, so a peer review other than by the code vendor was not possible.

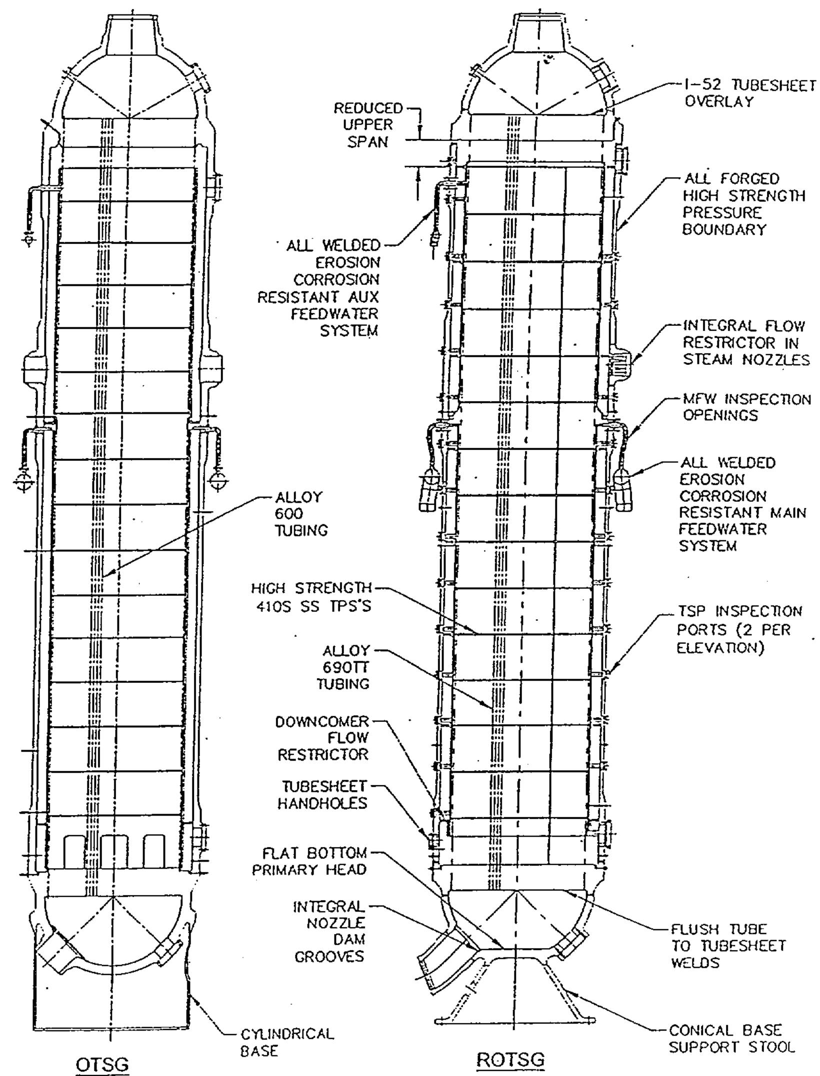
The Oconee replacement steam generators (ROTSGs) are being manufactured in Canada by Babcock & Wilcox Canada (BWC). The ROTSGs are quite similar functionally to the original OTSGs (Figure 1), and are characterized as a like-for-like replacement. The design differences that are important for thermal-hydraulic modeling are 1) flow-restricting orifices in the steam outlet nozzles, 2) Inconel-690 tubes, 3) 15,631 vs. 15,531 tubes, 4) thinner pressure vessel / wider downcomer, 5) thinner tubesheets resulting in 3.625 inch longer heated tube length, 6) 1.2% greater heat transfer area, and 7) more water in the steam generator. Of these design changes only the flow-restricting steam exit nozzles have a significant impact on the UFSAR Chapter 15 transient and accident analyses. These nozzles reduce the effective size of steam line breaks during blowdown of a steam generator to 1.804 ft^2. The other design changes will not cause a significant change in plant response during transients and accidents. For that reason the UFSAR Chapter 15 analysis methodology for the ROTSGs is similar to the modeling of the OTSGs, with changes limited to those described herein.

The Oconee UFSAR Chapter 15 non-LOCA transient analysis methodology through Revision 1 was based on the Electric Power Research Institute's RETRAN-02 code. The next generation in the RETRAN family of codes, RETRAN-3D, was approved by the NRC in the SER dated January 25, 2001. Revision 3 to DPC-NE-3000-P describes the modeling for Oconee with RETRAN-3D replacing RETRAN-02. [

] All of the details of the transition to using RETRAN-3D instead of RETRAN-02 for Oconee modeling are presented in DPC-NE-3000-P, Revision 3. Application of the Oconee RETRAN-3D model for analyzing the UFSAR Chapter 15 non-LOCA transients and accidents is detailed in this revision to DPC-NE-3005. The conditions and limitations in the NRC's SER for RETRAN-3D are also addressed.

The minor technical and editorial revisions are not necessarily associated with either the ROTSGs or the transition to RETRAN-3D. These changes are necessary to maintain the content of the report consistent with the current plant design and modeling, as well as to correct errors. No technical justification is necessary for editorial revisions and corrections. The changes are presented in the order that they appear in DPC-NE-3005. New Appendix A addresses the limitations and conditions in the RETRAN-3D SER as relates to the application to Oconee.

Figure 1 OTSG / ROTSG Comparison



Changes and Technical Justification

Cover Page and Frontal Pages

1. The revision and date will be revised

2. The table of contents and the lists of tables, figures, and acronyms will be updated

Chapter 1

3. Section 1.1, p. 1-1, second paragraph, correct a sentence that had a publishing error in Revision 1. (editorial)

   Change: "The application of these models for UFSAR Chapter 15 non-LOCA analyses for McGuire 1-11."

   To: "The application of these models for UFSAR Chapter 15 non-LOCA analyses for McGuire and Catawba was submitted to the NRC in References 1-10 and 1-11."

4. Section 1.1, p. 1-2, insert two new paragraphs at the bottom of the section to describe Revisions 1 and 2. (editorial)

   Insert the following two new paragraphs at the bottom:

   "Revision 1 was submitted in 1999 (Reference 1-33) in response to open issues in the NRC SER for Revision 0 (Reference 1-34). The main elements of the revision were, 1) a 110% of design pressure acceptance criterion for the locked rotor accident, 2) not credit operator action to trip the reactor following a SGTR, 3) not credit automatic main feedwater isolation for steam line break, 4) a new VIPRE-01 model for steam line break, 5) added the BWU-N CHF correlation for the Mk-B11 fuel assembly below the mixing vanes, and 6) added centerline fuel melt as a fuel damage criterion for the dropped rod analysis.

   Revision 2 includes methodology changes related to the replacement once-through steam generators, use of the RETRAN-3D code, minor technical methodology revisions, error corrections, and editorial changes."

5. Section 1.2, Computer Codes, p. 1-3, the RETRAN-02 code version used is updated to RETRAN-02 MOD5.2DKE. (editorial)

   Change: ". . . uses the RETRAN-02/MOD5.1 code, which . . . "

   To: ". . . uses the RETRAN-02MOD5.2 code, which is an error-corrected version of the RETRAN-02/MOD5.1 code that . . . "

6. Section 1.2, Computer Codes, p. 1-3, the RETRAN-3D code description is updated and the previous description is noted as being associated with the Revision 0 methodology. (editorial)

   Insert the following new text in front of the existing RETRAN-3D text:

   "RETRAN-3D: RETRAN-3D (Reference 1-19) was developed by Computer Simulation & Analysis, Inc. for EPRI to enhance and extend the simulation capabilities of the RETRAN-02 code. RETRAN-3D has many new and enhanced capabilities relative to RETRAN-02, in particular, a 3-D kinetics core model, improved two-phase models, an improved heat transfer correlation package, and an implicit numerical solution method. Most of the capabilities of the RETRAN-02 code have been retained within RETRAN-3D as options, except for a limited number of models and correlations that were not in use. The NRC SER for RETRAN-3D is dated January 25, 2001 (Reference 1-35). The SER includes limitations and conditions on the use of the code for licensing applications.

   The following paragraph is from the Revision 0 methodology, and is retained for historical purposes only:"

7. Section 1.3, p. 1-7, Credit for Control Systems and Non-Safety Components and Systems, Item 7, change this item due to a design modification. (plant design change)

   Change: "7) The turbine trip circuitry has two channels, one with a one second response time, and one with a fifteen second response time. The faster response time is credited in the current UFSAR Chapter 15 analyses and will be credited in the methodology. A station modification is planned to upgrade the second channel to a one second response time. The turbine trip circuitry is not completely safety-grade."

   To: "7) The turbine trip circuitry is not completely safety-grade."

   Technical Justification: A station modification has been installed that obtains a one second response time for both turbine trip channels. Therefore, the text describing the fifteen second response time can be deleted.

8. Section 1.3, p. 1-7, Credit for Control Systems and Non-Safety Components and Systems, Item 7, add a new item on post-trip main feedwater control. (design description)

   Insert the following at the bottom: "11) Control of the Main Feedwater System by the Integrated Control System is assumed to be in either the automatic or the manual mode of control initially, whichever is more conservative. The Integrated Control System transfers from manual to automatic control on reactor trip. Post-trip control is therefore in the automatic mode of control regardless of whether the event starts with the Main Feedwater System in manual or automatic control."

   Technical Justification: The methodology assumes that control systems respond as designed or remain in manual control, whichever assumption is more conservative. Item 11 is added to the text to clarify the modeling of post-trip main feedwater control. Main feedwater is assumed to be in either manual or automatic control, whichever is more conservative. However, if main feedwater is initially in manual control, then a severe overfeed of the steam generators will occur following any reactor trip. The Integrated

Control System is designed to transfer from manual to automatic control following a reactor trip to prevent overfeeding. That transfer is credited in the methodology.

9. Section 1.3, p. 1-7, Main Feedwater Isolation, change this item due to a design modification. (plant design change)

   Change: "The large and small steam line break analyses do not credit automatic isolation of main feedwater by the main steam line break detection and main feedwater isolation instrumentation."

   To: "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 & 3 (until AFIS is installed on those units)."

   Technical Justification: The AFIS design for automatic feedwater isolation has been installed on Unit 1 and will be installed on Units 2 & 3 in the next refueling outages. This revision is necessary to update the text to be consistent with the plant design. Since neither design is credited in the steam line break analysis methodology in this report, this change does not require NRC review.

10. New Section 1.4 "Interface with Duke Oconee Reload Design Methodology Topical Report" (editorial)

    1.4 <u>Interface with Duke Oconee Reload Design Methodology Topical Report</u>

    This report is referenced by NFS-1001A, "Duke Power Company Oconee Nuclear Station Reload Design Methodology" (Reference 1-2). Chapters 1 and 8 of NFS-1001A describe how the UFSAR Chapter 15 non-LOCA transient and accident analysis methodology of DPC-NE-3005 are integrated into the reload design process.

11. New Section 1.5 "Appendix A" (editorial)

    1.5 <u>Appendix A</u>

    Appendix A was added in Revision 2 to address the RETRAN-3D SER conditions and limitations as relates to the modeling for Oconee.

12. Section 1.5, References: References updated and new references added. (editorial)

    1-2 Duke Power Company Oconee Nuclear Station Reload Design Methodology, NFS-1001A, Revision 5, January 2001

    1-3 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002

    1-10 Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology, DPC-NE-3001-PA, Duke Power Company, December 2000

    1-11 FSAR Chapter 15 System Transient Analysis Methodology, DPC-NE-3002-A, Revision 3, Duke Power Company, May 1999

1-18 Mass and Energy Release and Containment Response Methodology, DPC-NE-3003-P, Revision 1, Duke Power Company, June 2002

1-19 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI, NP-7450(A), Volumes 1-4, Revision 5, July 2001

1-22 Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Revision 0, Duke Power Company, February 24, 1993

1-26 SIMULATE-3 Kinetics Theory and Model Description, SOA-96/26, Studsvik of America, April 1996

1-33 Letter, M. S. Tuckman (Duke) to NRC, February 1, 1999 (Letter submitting Revision 1 to DPC-NE-3005)

1-34 Letter, D. E. LaBarge (NRC) to W. R. McCollum (Duke), October 1, 1998 (NRC SER on DPC-NE-3005, Revision 0)

1-35 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

13. Renumbered Chapter 1 sections (editorial)

Chapter 2

14. Section 2.1, RETRAN-02/MOD5.1, p. 2-1, is updated to RETRAN-02/MOD5.2DKE, and Section 2.1.1, Code Description, is replaced to describe the current version of RETRAN-02. (model revision)

Change: "2.1 RETRAN-02MOD5.1"

To: "2.1 RETRAN-02MOD5.2DKE"

Change: " The RETRAN-02/MOD5.1 DKE code is a modified version of the NRC-approved RETRAN-02/MOD005.1 code (References 2-1 and 2-2). The version used in this topical report differs from the NRC-approved version in that two error corrections to the MOD5.1 version have been inserted to obtain the MOD5.1DKE version. These two error corrections, which will be included in future versions of the RETRAN code are:

- Parameter LPOOL in subroutine INPUT was changed from 125,000 to 175,000. This modification was necessary to allow a large RETRAN input deck to execute.

- The DO statement in loop 920 of subroutine GENTRN is changed from "DO 920 IGS = 1,10" to "DO 920 IGS = 1,30". This correction enhances convergence in the generalized transport model.

Since the RETRAN version used is NRC-approved with the exception of error corrections, it is concluded that this report need not justify the validity of the RETRAN code itself. Therefore, details regarding the theory of the RETRAN code are left to the references."

To: "The RETRAN-02/MOD5.2DKE code is a modified version of the NRC-approved RETRAN-02/MOD005.1 code (References 2-1 and 2-2). The EPRI released MOD5.2 version contains user features and error corrections to the NRC-approved MOD 5.1 version - but no new models. The MOD5.2 version used in this topical report differs from the EPRI-released version in that three additional changes to the MOD5.2 version have been made to obtain the MOD5.2DKE version. These three changes are as follows:

- Parameter LPOOL was changed from 200,000 to 400,000 to allow larger problems to run. This modification was necessary only to allow a large RETRAN input deck to execute.

- A custom code modification was added to allow access to the condensation heat transfer correlations with the forced convection heat transfer map. This was necessary since the forced convection heat transfer map is used, and condensation heat transfer will occur during some applications.

- An error in the calculation of the liquid region work term in the pressurizer model was corrected.

The first change is a problem execution change only. The second change was described in the Reference 2-3 methodology that has been approved by the NRC. The third change is an error correction. None of these changes require NRC review."

Technical Justification: RETRAN-02MOD5.1 was the version of RETRAN-02 used at the time of submittal of the original version of this report. RETRAN-02MOD5.1 was reviewed and approved by the NRC. RETRAN-02MOD5.2 contains only user features and error corrections (no new models) relative to the NRC-approved MOD5.1 version. For that reason NRC review of MOD5.2 was not necessary. Three changes have been made to the EPRI-released version of RETRAN-02MOD5.2 to create the Duke version RETRAN-02MOD5.2DKE. The first change (parameter LPOOL increased to a value of 400,000) is necessary to run large problems. This change is an execution change only, and is not a model revision. The second change is to allow access to the condensation heat transfer correlations when using the forced convection heat transfer map. This custom coding change was previously described in Reference 2-3, which has been reviewed and approved by the NRC. The third change is an error correction made by the code vendor in response to investigating an apparent error. Code error corrections do not require NRC review and approval. Therefore, there are no new models associated with Duke version RETRAN-02 MOD5.2DKE, and no NRC review is necessary.

15. Section 2.2, RETRAN-3D/MOD001F, p. 2-9, is updated to RETRAN-3D/MOD3.1DKE. Section 2.2.1, Code Description, is replaced to describe the version of RETRAN-3D used by Duke. Section 2.2.2, Simulation Model, is revised to refer to Reference 2-3. Section 2.2.3, Validation of Code and Model, is revised to refer to Reference 2-3. (model revision)

Change: "2.2 RETRAN-3DMOD001F"

To: "2.2 RETRAN-3DMOD3.1DKE"

Change: "2.2.1 Code Description The RETRAN-3D/MOD001F code (Reference 2-7) is a recent version of the EPRI RETRAN family of codes. RETRAN-3D was developed to provide analysis capabilities for LWR transients, small-break loss-of-coolant accidents, anticipated transients without scram, natural circulation events, long-term transients, and transients with thermodynamic non-equilibrium phenomena. New fully implicit numerical solution schemes have been provided for both fluid states, including additional balance equations to better predict non-equilibrium phenomena. The code user has the ability to run RETRAN-3D in several modes. The application of RETRAN-3D in this report is limited to the "RETRAN-02 mode", which does not include any of the non-equilibrium, flowing non-condensible gas capability, or three-dimensional core modeling unique to RETRAN-3D. The advanced solution scheme and correlations are used in the analyses in this report."

To: "2.2.1 Code Description RETRAN-3D (Reference 2-7) was developed by Computer Simulation & Analysis, Inc. for EPRI to enhance and extend the simulation capabilities of the RETRAN-02 code. RETRAN-3D has many new and enhanced capabilities relative to RETRAN-02, in particular, a 3-D kinetics core model, improved two-phase models, an improved heat transfer correlation package, and an implicit numerical solution method. Most of the capabilities of the RETRAN-02 code have been retained within RETRAN-3D as options, except for a limited number of models and correlations that were not in use. The only new RETRAN-3D model used in the Revision 2 methodology is in [

] All of the details of the transition to using RETRAN-3D instead of RETRAN-02 for Oconee modeling are presented in DPC-NE-3000-P, Revision 3 (Reference 2-3). The NRC SER for RETRAN-3D is dated January 25, 2001 (Reference 2-23). The SER includes limitations and conditions on the use of the code for licensing applications. Reference 2-3 describes the Duke code modifications to the EPRI-released version. Appendix A of this report addresses compliance with the SER limitations and conditions."

Change: "2.2.2 Simulation Model The Oconee RETRAN Model discussed in Section 2.1.2 is used with the RETRAN-3D/MOD001F code. Appropriate input deck changes are made to account for the relocation of certain parameters in the RETRAN-3D input deck, including additional input requirements. The transient-specific modeling revisions/additions required for each transient modeled with RETRAN-3D are the same as those made for the RETRAN-02 analysis. Other than these modifications, the RETRAN model used for this report is the same as that reviewed by the NRC per Reference 2-3."

To: "2.2.2 Simulation Model The Oconee RETRAN model is described in Reference 2-3.

Change: "Validation of Code and Model   Validation of the Oconee RETRAN Model for transient analysis is documented in References 2-3 and 2-4 for use with the RETRAN-02 code. Four of the transients simulated with RETRAN-02 in this topical report are also simulated with the RETRAN-3D code. The RETRAN-3D code is run in the "RETRAN-02 mode" to show that application of RETRAN-3D gives very similar results to the NRC-approved RETRAN-02 code. Generic validation of RETRAN-3D was performed by EPRI and is documented in the RETRAN-3D manuals (Reference 2-7).

To: "2.2.3   Validation of Code and Model   Validation of the code and model is described in References 2-3 and 2-7.

Technical Justification: The NRC-approved RETRAN-3DMOD3.1 code will be used for applications with the replacement steam generators. The Duke modifications to the MOD3.1 version, the Oconee simulation model, Duke validation, and compliance with the limitations and conditions of the SER are all detailed in Reference 2-3. Generic validation of the RETRAN-3D code is presented in Reference 2-7.

16. Section 1.5, References: References updated and one new reference added. (editorial)

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

    2-7 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI, NP-7450(A), Volumes 1-4, Revision 5, July 2001

    2-10 Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology, DPC-NE-3001-PA, Duke Power Company, December 2000

    2-15 Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Revision 0, Duke Power Company, February 24, 1993

    2-23 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

<u>Chapter 3</u>

17. Section 3.5, References: Reference 3-1 updated. (editorial)

    3-1 Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Revision 0, Duke Power Company, February 24, 1993

<u>Chapter 4</u>

18. Section 4.6, References, Reference 4-1 updated. (editorial)

4-1 Thermal-Hydraulic Statistical Core Design Methodology, DPC-NE-2005P-A, Revision 2, Duke Power Company, April 17, 2000

Chapter 5

19. Section 5.5, References, Reference 5-3 updated. (editorial)

5-3 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002

Chapter 6

20. Section 6.5, p. 6-11, fifth bullet says that we check each reload core for the maximum allowable radial peak limits. This will be deleted since this event is non-limiting. (model revision)

Delete: " o Maximum allowable radial peak limits"

Technical Justification: Experience with analyzing core reloads since the original submittal of this report has shown that the rod withdrawal at power transient is non-limiting with regard to the DNBR limit. The purpose of checking the maximum allowable radial peak limits is to confirm that DNBR margin exists for this event. Based on experience, there is no possibility that a negative margin situation will occur in the future for this event without being identified for the limiting DNBR events. Therefore, the resources to perform this check during the reload process are wasted. This methodology revision is appropriate since deleting this check has no technical significance and resources will be saved.

21. Section 6.6, References, Reference 6-4 updated. (editorial)

6-4 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002

Chapter 8

22. Section 8.8, References, Reference 8-2 updated. (editorial)

8-2 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002

Chapter 9

23. Section 9.3.2.5, p. 9-9 is revised to clarify that only the two-pump coastdown from four pumps in operation (not also the one-pump coastdown from three pumps in operation statepoint) statepoint is used in the determination of core power peaking limits. (editorial)

Change: "These statepoints are then used . . . "

To: "The two-pump coastdown from four pumps in operation statepoint is then used . . ."

24. Section 9.4, References, references updated. (editorial)

9-2 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002

9-5 The BWU Critical Heat Flux Correlations, BAW-10199P-A, Framatome Advanced Nuclear Products, August 1996

9-6 Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Duke Power Company, February 24, 1993

9-7 Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2003P-A, Revision 1, Duke Power Company, September 2000

Chapter 10

25. Section 10.5, References, references updated. (editorial)

10-2 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002

10-5 The BWU Critical Heat Flux Correlations, BAW-10199P-A, Framatome Advanced Nuclear Products, August 1996

10-6 Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Duke Power Company, February 24, 1993

Chapter 11

26. Section 11.1.5, Control, Protection, and Safeguards Systems, Reactor Trip, the flux/flow/imbalance trip was inadvertently missing from the text. (editorial)

Change: "The reactor trip functions credited are the high flux trip and the high RCS pressure trip."

To: "The reactor trip functions credited are the high flux trip, the flux/flow/imbalance trip, and the high RCS pressure trip."

27. Section 11.1.5, Control, Protection, and Safeguards Systems, Excore Flux Instrumentation, a new methodology is proposed to model the effect of a change in reactor vessel downcomer water temperature on the excore flux detector signal. (model revision)

Insert the following new text at the bottom:

"The effect of a change in reactor vessel downcomer water temperature on the excore flux detector signal is based on one of two methods. The first method, used through Revision 1 of the methodology, is a synthesis of plant data and analysis results obtained from Framatome Advanced Nuclear Products. The second method, submitted with Revision 2 of the methodology, is described in the following paragraphs. Either method may be used in the application of the methodology.

The effect of Reactor Coolant System temperature on the excore power range neutron detectors is modeled by consideration of the relevant source, material composition, and detector geometry details. The fuel region is considered as a homogeneous mixture inside a cylinder of equivalent volume as the core region inside the baffle plates. The balance of the geometry is modeled as a series of concentric cylinders, representing the baffle plates, flow channel, core barrel, thermal shield, downcomer region, and reactor vessel. Particle tallies at the detector account for the details of detector geometry and design.

Fuel characterization is performed with the SAS2H/ORIGEN-S modules of the SCALE Code System, as necessary (References 11-5 and 11-6). Transport and tallying of particles is performed with the MCNP computer code (Reference 11-7). Variance reduction is performed in MCNP as necessary to achieve reliable statistics for the tallies. The particle tally results are used to characterize the relative effects of Reactor Coolant System temperature on detector response."

Technical Justification: The methodology models the effect of a reduction in the reactor vessel downcomer water temperature on the excore flux detector. This can be a significant effect during overcooling transients, since the detector will indicate a lower power level than the actual core power level. The modeling of this effect through Revision 1 has involved a synthesis of plant data and code predictions obtained from Framatome Advanced Nuclear Products. This revision to the methodology proposes an updated analytical method using the SAS2H/ORIGEN-S modules of the SCALE Code System and the MCNP Monte-Carlo N-particle transport code. The method will be used to calculate the magnitude of the change in the excore flux signal for a given change in reactor vessel downcomer water temperature. This will then be modeled in RETRAN to ensure that the flux detector signal and the resulting reactor trip signal are appropriately addressing this effect.

28. Section 11.3, References, Reference 11-4 updated, and three new references are added. (editorial)

11-4 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002

11-5 O. W. Hermann and C. V. Parks, "SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Module," NUREG/CR-0200, Volume 1, Section S2.

11-6 O. W. Hermann and R. M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," NUREG/CR-0200, Volume 2, Section F7.

11-7 Judith F. Briesmeister, Ed., "MCNP – A General Monte Carlo N-Particle Transport Code," Los Alamos National Laboratory Report, LA-13709-M.

Chapter 12

29. Section 12.5, References, references updated. (editorial)

12-2 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002

12-3 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI, NP-7450(A), Volumes 1-4, Revision 5, July 2001

Chapter 13

30. Section 13.3, Boundary Conditions, Manual Operator Actions, p. 13-5, the first bullet is moved up to be the fourth bullet on p. 13-4. This is necessary to keep the action in the correct time sequence. (editorial)

Insert the first bulleted item on p. 13-5 after the third bulleted item on p. 13-4.

31. Section 13.3, Boundary Conditions, Manual Operator Actions, p. 13-4, the fourth bullet is revised for consistency with the Oconee emergency operating procedures. (model revision)

Change: "An operator action delay time is assumed from the time the ruptured steam generator is identified to the time that a cooldown of the RCS to 532°F begins."

To: "An operator action delay time is assumed from the time RCS subcooled margin is minimized to the time that a cooldown of the RCS to 532°F begins "

Technical Justification: The Oconee emergency operating procedures are periodically revised to improve the accident mitigation and recovery guidance. Consequently it is necessary to periodically revise the analysis methodology so that the operator actions credited in the analysis remain consistent with the station procedures. This is particularly necessary for the steam generator tube rupture accident since it involves a large number of manual operator actions. In this revision the time delay prior to starting the unit cooldown does not change, but the action that keys the start of the cooldown does. Previously that action was the time of identification of the ruptured steam generator. With this revision that action is the completion of the minimizing of the RCS subcooled margin.

32. Section 13.3, Boundary Conditions, Manual Operator Actions, p. 13-4, the fifth bullet is revised for consistency with the Oconee emergency operating procedures. (model revision)

Change: "An operator action delay time of 10 minutes after reaching 532°F is assumed."

To: "An operator action delay time of 20 minutes after reaching 532°F is assumed "

Technical Justification: The Oconee emergency operating procedures are periodically revised to improve the accident mitigation and recovery guidance. Consequently it is necessary to periodically revise the analysis methodology so that the operator actions credited in the analysis remain consistent with the station procedures. This is particularly necessary for the steam generator tube rupture accident since it involves a large number of manual operator actions. In this revision the time delay prior to isolating the ruptured steam generator is increased from 10 to 20 minutes. This increase is necessary based on procedure validation data that indicated that more time was necessary in the analysis assumption to bound the observed crew response time.

33. Section 13.3, Boundary Conditions, Manual Operator Actions, p. 13-5, the second bullet is revised for consistency with the Oconee emergency operating procedures. (model revision)

Change: "One RCP per loop is tripped off after the RCS has cooled down to 532°F. A 10 minute operator action delay time is assumed for this after reaching 532°F."

To: "One RCP in the loop without the pressurizer is tripped off after the RCS has cooled down to 532°F. A 20 minute operator action delay time is assumed for this after reaching 532°F. One RCP in the loop with the pressurizer is tripped at 400°F."

Technical Justification: The Oconee emergency operating procedures are periodically revised to improve the accident mitigation and recovery guidance. Consequently it is necessary to periodically revise the analysis methodology so that the operator actions credited in the analysis remain consistent with the station procedures. This is particularly necessary for the steam generator tube rupture accident since it involves a large number of manual operator actions. In this revision the procedural guidance for stopping the reactor coolant pumps has been revised. The first revision clarifies that the first pump to be stopped is to be in the loop without the pressurizer. This will maximize the pressurizer spray capability. The second revision is that the operator action delay is increased from 10 to 20 minutes. This increase is necessary based on procedure validation data that indicated that more time was necessary in the analysis assumption to bound the observed crew response time. The third revision clarifies that the second pump to be stopped will be in the opposite loop to obtain a one/one operating configuration. This action is to be taken at 400°F. Stopping the second pump reduces the heat load and allows a faster cooldown rate. The one/one pump operating configuration is desired for positive flow in both loops and continued pressurizer spray capability.

34. Section 13.3, Boundary Conditions, Manual Operator Actions, p. 13-5, the fifth bullet is revised for clarification purposes. (editorial)

Change: "The RCS boron concentration is determined at this time."

To: "RCS boron sampling is completed at this time."

35. Section 13.3, Boundary Conditions, Manual Operator Actions, p. 13-5, the last bullet is revised for consistency with the Oconee emergency operating procedures. (model revision)

Change: "An operator action delay time of 45 minutes is assumed."

To: "An operator action delay time of 90 minutes is assumed."

Technical Justification: The Oconee emergency operating procedures are periodically revised to improve the accident mitigation and recovery guidance. Consequently it is necessary to periodically revise the analysis methodology so that the operator actions credited in the analysis remain consistent with the station procedures. This is particularly necessary for the steam generator tube rupture accident since it involves a large number of manual operator actions. In this revision the time delay prior to align the decay heat removal system is increased from 45 to 90 minutes. This increase is necessary based on procedure validation data that indicated that more time was necessary in the analysis assumption to bound the observed crew response time.

36. Section 13.8, References, Reference 13-1 updated. (editorial)

13-1 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002

Chapter 14

37. Section 14.6, References, Reference 14-9 updated. (editorial)

14-9 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002

Chapter 15

38. Section 15.2.1.3, Steam Generator Model, p. 15-4, a note is added to indicate that the text in this section related to inputs and modeling choices is applicable to the RETRAN-02 analysis for the original steam generators. (model revision)

Insert the following note at the beginning:

"Note: The inputs and modeling techniques in the section are applicable to the RETRAN-02 analysis for the original steam generators."

Technical Justification: This section describes inputs and modeling techniques that were used for the steam line break analysis using RETRAN-02 for the original steam generators. Some of these are not applicable to the RETRAN-3D analysis for the replacement steam generators. For examples, zero steam generator tube plugging is assumed for the replacement steam generators; the isoenthalpic expansion choked flow option is only used in Junctions 134 and 234 (wasn't needed in the other junctions); and the [ ] used. Since Revision 2 does not intend to include all of the inputs and modeling techniques that are not applicable to the RETRAN-3D analysis for the replacement steam generators.

39. Section 15.2.1.6, Break Model, p. 15-6, add text to discuss the flow restricting orifices in the steam outlet nozzles that are part of the replacement steam generator design. (plant design change).

Insert a new second paragraph:

"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."

Technical Justification: The replacement steam generators have been designed with flow restricting orifices in the steam outlet nozzles to reduce the blowdown rate and consequences. These orifices are modeled in RETRAN and have the effect of reducing the break flow after the initial depressurization of the main steam lines.

40. Section 15.3.1.1.4, Control, Protection, and Safeguards Systems, p. 15-15, replace "Main Steam Line Break Detection and Main Feedwater Isolation Instrumentation" with "Feedwater Isolation" to include the Automatic Feedwater Isolation System design modification. (plant design change)

Insert the following title and paragraph:

"<u>Feedwater Isolation</u> "The steam line break analyses do not credit automatic isolation of 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 & 3 (until AFIS is installed on those units)."

Technical Justification: The AFIS design for automatic feedwater isolation has been installed on Unit 1 and will be installed on Units 2 & 3 in the next refueling outages. This revision is necessary to update the text to be consistent with the plant design. Since neither design is credited in the steam line break analysis methodology in this report, this change does not require NRC review.

41. Section 15.3.2.1.4, Control, Protection, and Safeguards Systems, p. 15-22, replace "Main Steam Line Break Detection and Main Feedwater Isolation Instrumentation" with "Feedwater Isolation" to include the Automatic Feedwater Isolation System design modification. (plant design change)

Insert the following title and paragraph:

"<u>Feedwater Isolation</u> "The steam line break analyses do not credit automatic isolation of 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 & 3 (until AFIS is installed on those units)."

Technical Justification: The AFIS design for automatic feedwater isolation has been installed on Unit 1 and will be installed on Units 2 & 3 in the next refueling outages. This revision is necessary to update the text to be consistent with the plant design. Since neither design is credited in the steam line break analysis methodology in this report, this change does not require NRC review.

42. Section 15.5, References, references updated. (editorial)

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

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

Chapter 16

43. Section 16.1.3, Boundary Conditions, Excore Flux Detector Error Due To Overcooling, a new methodology is proposed to model the effect of a change in reactor vessel downcomer water temperature on the excore flux detector signal. (model revision)

Insert the following new text at the bottom:

"The effect of a change in reactor vessel downcomer water temperature on the excore flux detector signal is based on one of two methods. The first method, used through Revision 1 of the methodology, is a synthesis of plant data and analysis results obtained from Framatome Advanced Nuclear Products. The second method, submitted with Revision 2 of the methodology, is described in the following paragraphs. Either method may be used in the application of the methodology.

The effect of Reactor Coolant System temperature on the excore power range neutron detectors is modeled by consideration of the relevant source, material composition, and detector geometry details. The fuel region is considered as a homogeneous mixture inside a cylinder of equivalent volume as the core region inside the baffle plates. The balance of the geometry is modeled as a series of concentric cylinders, representing the baffle plates, flow channel, core barrel, thermal shield, downcomer region, and reactor vessel. Particle tallies at the detector account for the details of detector geometry and design.

Fuel characterization is performed with the SAS2H/ORIGEN-S modules of the SCALE Code System, as necessary (References 16-4 and 16-5). Transport and tallying of particles is performed with the MCNP computer code (Reference 16-6). Variance reduction is performed in MCNP as necessary to achieve reliable statistics for the tallies. The particle tally results are used to characterize the relative effects of Reactor Coolant System temperature on detector response."

Technical Justification: The methodology models the effect of a reduction in the reactor vessel downcomer water temperature on the excore flux detector. This can be a significant effect during overcooling transients, since the detector will indicate a lower power level than the actual core power level. The modeling of this effect through Revision 1 has involved a synthesis of plant data and code predictions obtained from Framatome Advanced Nuclear Products. This revision to the methodology proposes an updated analytical method using the SAS2H/ORIGEN-S modules of the SCALE Code System and the MCNP Monte-Carlo N-particle transport code. The method will be used to calculate the magnitude of the change in the excore flux signal for a given change in reactor vessel downcomer water temperature. This will then be modeled in RETRAN to ensure that the flux detector signal and the resulting reactor trip signal are appropriately addressing this effect.

44. Section 16.1.5, Control, Protection, and Safeguards Systems, p. 16-5, replace "Main Steam Line Break Detection and Main Feedwater Isolation Instrumentation" with "Feedwater Isolation" to include the Automatic Feedwater Isolation System design modification. (plant design change)

Insert the following title and paragraph:

"Feedwater Isolation "The steam line break analyses do not credit automatic isolation of 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 & 3 (until AFIS is installed on those units)."

Technical Justification: The AFIS design for automatic feedwater isolation has been installed on Unit 1 and will be installed on Units 2 & 3 in the next refueling outages. This revision is necessary to update the text to be consistent with the plant design. Since neither design is credited in the steam line break analysis methodology in this report, this change does not require NRC review.

45. Section 16.5, References, Reference 16-3 updated and three new references added. (editorial)

16-3 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002

16-4 O. W. Hermann and C. V. Parks, "SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Module," NUREG/CR-0200, Volume 1, Section S2.

16-5 O. W. Hermann and R. M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," NUREG/CR-0200, Volume 2, Section F7.

16-6 Judith F. Briesmeister, Ed., "MCNP – A General Monte Carlo N-Particle Transport Code," Los Alamos National Laboratory Report, LA-13709-M.

APPENDIX A

EVALUATION OF RETRAN-3D SER CONDITIONS AND LIMITATIONS FOR THE OCONEE RETRAN MODEL WITH ROTSGs

Purpose

This appendix evaluates the conditions and limitations in the RETRAN-3D SER (Reference A-1) for the application of RETRAN-3D to the Oconee Nuclear Station with replacement steam generators (ROTSGs). The results of this evaluation demonstrate that the use of the RETRAN-3D code for this application has been appropriately justified and is within the SER conditions and limitations. Therefore, the approval for use of RETRAN-3D as stated in the SER for the scope of approval specified in the SER can be credited. The Duke version of the RETRAN-3D code actually used is described first, including details on the custom code modifications that have been incorporated.

Description of the RETRAN-3D Code Version Used and Duke Code Modifications

RETRAN-3D MOD003.1DKE is the current Duke Power version of the standard Electric Power Research Institute (EPRI) RETRAN-3D MOD003.1 code, the NRC-approved code. This EPRI version includes the revisions agreed to during the SER review process. The Duke Power version, designated by the suffix "DKE", consists of two types of revisions to the standard EPRI version. The first type of revision is code error corrections. Duke periodically updates the code version in use to include error corrections obtained from Computer Simulation & Analysis, Inc. (CSA), the EPRI contractor for the RETRAN-3D code. The second type of revision is Duke custom code modifications purchased from CSA to address Duke-specific modeling needs. Each of these custom code changes are described in detail for NRC review and approval. The error corrections are not presented. All error corrections and code modifications are developed and controlled under CSA's Appendix B QA program. All Duke RETRAN-3D versions used for safety-related applications are certified and controlled under Duke's Appendix B software quality assurance program (Reference A-2). Any future modifications to the RETRAN-3D code versions used by Duke for safety-related applications that constitute significant model revisions or new models will be submitted for NRC review and approval. Code modifications that consist of error corrections or user features will be implemented under QA processes, but will not be submitted for NRC review.

Duke Code Modification #1 — Allow Access to the Condensation Heat Transfer Correlations With the Forced Convection Heat Transfer Map

Initialization of the ROTSGs using RETRAN-3D uses the forced convection heat transfer map. This is selected by setting variable IHTMAP on the 01000Y card to a value of zero. This standard forced convection option does not allow access to the condensation heat transfer correlations in RETRAN-3D. A code modification was implemented to allow access to the

condensation heat transfer correlations by setting IHTMAP to a value of 2. This option gives the forced heat transfer map, but allows condensation heat transfer to be modeled when appropriate for the local conditions present. This can be important in several transient conditions, such as when the primary water flowing through the steam generator tubes cools to below the secondary saturation temperature. In this situation an appropriate condensation heat transfer coefficient will be selected from the heat transfer correlation set. The technical justification for this code modification is that it allows a correct heat transfer correlation to be used for situations when condensation heat transfer occurs. This modification is functionally equivalent to a similar update made to RETRAN-02, which has been reviewed and approved for use by the NRC.

Duke Code Modification #2 — Allow the User to Specify the Dittus-Boelter Heat Transfer Correlation for a Specific Conductor

The RETRAN-3D heat transfer correlation package selects an appropriate heat transfer correlation for each conductor surface based on fluid conditions in adjacent volumes. Under some conditions it is useful to be able to select a specific correlation for a given conductor surface rather than using the code-selected correlation. This code modification allows the user to specify either the Dittus-Boelter liquid or vapor correlation for the left surface of a particular conductor. This will then override the code-selected correlation. Word IMCL on Card 15XXXY is set to a value of 41 to specify the liquid correlation, and to a value of 48 to specify the vapor correlation at the left surface of a conductor. The need for this modeling capability arose during analyses of the ROTSG upper shell heat transfer following a steam line break. Due to water carryout into the steam outlet annulus (the volume adjacent to the upper shell), the heat transfer was potentially too high for an analysis in which less heat transfer was conservative. This code modification allowed specifying use of the Dittus-Boelter correlation for vapor to conservatively model the heat transfer to the shell conductor. The code modification also allows specification of the Dittus-Boelter correlation for liquid as another modeling option. The technical justification for this code modification is that a capability to specify a heat transfer correlation for specific applications is appropriate.

Duke Code Modification #3

Duke Code Modification #4

Duke Code Modification #5

Evaluation of RETRAN-3D SER Conditions and Limitations

1. Multidimensional neutronic space-time effects cannot be simulated as the maximum number of dimensions is one. Conservative usage has to be demonstrated.

Staff Position: RETRAN-3D has been modified to include a 3-dimensional nodal kinetics model based on the analytic nodalization method similar to accepted codes. The code has been assessed by calculation of the response of the SPERT prompt-critical tests and has been confirmed by the staff by comparisons with calculations performed with the NESTLE and TORT codes. The staff concludes that the code can adequately predict the response to prompt-critical events such as the PWR rod ejection accident and the BWR rod drop accident. If void generation occurs from an initially un-voided case, the user will have to justify crediting this negative feedback in the analysis.

The code was used by a participant in the Nuclear Energy Agency's International Standard Problem calculation of a hypothetical main steam line break (MSLB) at the Three Mile Island Unit 1 plant. The results of the calculation comparison indicates that RETRAN-3D is comparable to any of the other participating codes.

RETRAN-3D is approved for main steam line break analyses subject to the following conditions. Thermal-hydraulic effects can have a large impact on the cross section evaluation and thus on the resulting power distribution and magnitude. Therefore, the licensee must justify the primary side nodalization for mixing in the vessel and core. The licensee must also evaluate the uncertainties in the modeling.

Duke Position: The RETRAN-3D three-dimensional kinetics model is not used.

2. There is no source term in the neutronics and the maximum number of energy groups is two. The space-time options assume an initially critical system. Initial conditions with zero fission power cannot be simulated by the kinetics. The neutronic models should not be started from subcritical or with zero fission power without further justification.

Staff Position: The basic models in RETPAN-3D are unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not analyze from subcritical or zero fission power initial conditions.

3. A boron transport model is unavailable. User input models will have to be reviewed on an individual basis.

Staff Position: As noted previously in this report, boron transport is handled as a "contaminant" by the "general transport model." This model uses first order accurate upwind difference scheme with an implicit temporal differencing. This approach is well known for being highly diffusive, especially if the Courant limit is exceeded. Since RETRAN-3D has the same model as RETRAN-02 MODOO3 and subsequent versions that have been approved for use, the RETRAN-3D model is also approved with the caveat that the potential to produce misleading results with this scheme necessitates careful review of the results for any case where boron transport/dilution is important.

Duke Position: Previously approved in RETRAN-02 and for Duke applications using RETRAN-02.

4. Moving control rod banks are assumed to travel together The BWR plant qualification work shows that this is an acceptable approximation.

Staff Position: The control bank limitation is applied only to the one-dimensional kinetics model. The staff agrees that the 3-dimensional kinetics model need not be restricted in this way.

Duke Position: Resolved per the Staff Position

5. The metal-water heat generation model is for slab geometry The reaction rate is therefore under-predicted for cylindrical cladding. Justification will have to be provided for specific analyses.

Staff Position: The basic models in RETRAN-3D are unchanged and, therefore, this condition of use applies. However, since RETRAN-3D is not being reviewed for loss-of-coolant accident analysis, where core uncovery and heatup are significant, this condition does not occur in the transients for which application of RETRAN-3D has been reviewed.

Duke Position: Duke does not use the metal-water heat generation model.

6. Equilibrium thermodynamics is assumed for the thermal-hydraulics field equations although there are nonequilibrium models for the pressurizer and the subcooled boiling region.

Staff Position: The RETRAN-3D five equation model permits thermal-hydraulic nonequilibrium between the liquid and vapor phases. While it allows subcooled liquid and saturated steam to be concurrently present, it does not account for subcooled liquid and superheated vapor being concurrently present. Use of the code is not approved for LOCA. Also, the user must be aware of this limitation and avoid conditions which will place subcooled liquid and superheated vapor in contact.

Duke Position: Duke does not use the RETRAN-3D five equation model.

7. While the vector momentum model allows the simulation of some vector momentum flux effects in complex geometry the thermal-hydraulics are basically one-dimensional.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this comment still applies.

Duke Position: As described in Appendix B, Duke is proposing to use the vector momentum option with junction angle input for certain junctions where the momentum flux terms are considered to be potentially important. It is acknowledged that the thermal-hydraulics are basically one-dimensional. Duke's use of this model is based on vendor recommendations.

8. Further justification is required for the use of the homogeneous slip options with BWRs.

Staff Position: RETRAN-3D has five slip equation options for the user to choose from, three of which are retained from RETRAN-02 for compatibility. The recommended model options are based on the Chexal-Lellouche drift flux correlation. The first is the algebraic slip model, which

is approved for use with BWR bundle geometry as given in condition (9). The second is a form of the dynamic slip model that uses the Chexal-Lellouche drift flux correlation to evaluate the interfacial friction approved in condition (10). The user must justify the use of any other slip options.

Duke Position: Duke is not modeling BWRs.

9. The drift flux correlation used was originally calibrated to BWR situations and the qualification work for both this option and for the dynamic slip option only cover BWRs. The drift flux option can be approved for BWR bundle geometry if the conditions of (16) are met.

Staff Position: The Chexal-Lellouche drift flux model has been used in comparisons with FRIGG-2 and FRIGG-4 void fraction data and is acceptable for use in BWR bundle geometry.

Duke Position: Duke is not modeling BWRs.

10. The profile effect on the interphase drag (among all the profile effects) is neglected in the dynamic slip option. Form loss is also neglected for the slip velocity. For the acceptability of these approximations refer to (17).

Staff Position: Form loss terms have been included in the RETRAN-3D dynamic slip model. The Taugl form of the dynamic slip equation also includes profile effects in the interphase drag model. These RETRAN-3D model improvements adequately address the concerns and the model is approved for use when the Chexal-Lellouche model is used to compute the interphase friction. Approval is subject to the conditions given in (16) for the Chexal-Lellouche drift flux correlation. Users must justify use of any other dynamic slip option.

Duke Position: Duke is not using the dynamic slip option.

11. Only one-dimensional heat conduction is modeled. The use of the optional gap linear thermal expansion model requires further justification.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use still applies.

Duke Position: Duke is not using the optional linear gap thermal expansion model.

12. Air is assumed to be an ideal gas with a constant specific heat representative of that at containment conditions. It is restricted to separated and single-phase vapor volumes. There are no other noncondensables.

Staff Position: RETRAN-3D has been extended to include a general noncondensable gas capability which resolves the original concern. However, the noncondensable gas flow model is approved for use subject to the following restriction.

As noted in Section 111.3.0 of the RETRAN-3D Theory Manual (Reference 4), none of the models available for calculating critical flow are appropriate when noncondensable gases are present. Consequently, the code automatically bypasses the critical flow model when noncondensable gases are present in a junction. Users must confirm that noncondensable flows do not exceed appropriate critical flow values or justify use of values that may exceed critical flow values.

<u>Duke Position</u>: Duke is not using the noncondensable gas flow model.

13. The use of the water properties polynomials should be restricted to the subcritical region. Further justification is required for other regions.

<u>Staff Position</u>: For enthalpies less than approximately 820 Btu/lbm, the difference between the ASME and RETRAN-3D curve fit values of the specific volume range from less than 0.2 percent to approximately 1.3 percent for pressures ranging from 0.1 to 6,000 psia. Further, for enthalpies greater than 820 Btu/lbm and pressures greater than 4200 psia, the differences in specific volume are also less than 1.0 percent. RETRAN-3D is approved for use with PWR ATWS analyses where the peak pressure resides in the regions described above.

For enthalpies greater than 820 Btu/lbm and pressures between 3200 and 4200 psia, the differences in specific volume increase as the enthalpy increases and the pressure decreases. The maximum error of approximately 3.8 percent occurs at the critical point. PWR ATWS analysis using RETRAN-3D in this region will require additional justification that the difference in specific volume does not adversely affect the calculation of the peak pressure.

<u>Duke Position</u>: Duke will address the above condition if an application encounters conditions in the region of concern.

14. A number of regime-dependent minimum and maximum heat fluxes are hardwired. The use of the heat transfer correlations should be restricted to situations where the pre-CHF heat transfer or single-phase heat transfer dominates.

<u>Staff Position</u>: RETRAN-3D contains both the "forced convection option" contained in RETRAN-02 which is the basis for this restriction, and a second option referred to as the "combination heat transfer map." If the first option is chosen, the "forced convection option," approval is granted only for use in pre-CHF and single-phase heat transfer regimes. If the second option is chosen, the "combination heat transfer map," then there are no discontinuities between successive heat transfer regimes and the appropriate heat transfer value should result. Therefore, the combination heat transfer option is approved for use.

<u>Duke Position</u>:

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15. The Bennett flow map should only be used for vertical flow within the conditions of the data base and the Beattie two-phase multiplier option requires qualification work

Staff Position: The Beattie two-phase multiplier has been removed from RETRAN-3D. The Govier horizontal flow map has been added to supplement the Bennett map for vertical flow and is acceptable.

Duke Position: Resolved per the Staff Position.

16. No separate effects comparison have been presented for the algebraic slip option and it would be prudent to request comparisons with the FRIGG tests before the approval of the algebraic slip option.

Staff Position: The algebraic slip option has been modified to include the Chexal-Lellouche drift flux model. Use of the Chexal-Lellouche drift flux model for BWR and PWR applications within the range of conditions covered by the steam-water database used to develop and validate the model is approved. The model has been qualified with data from a number of steady-state and two-component tests. While the small dimensions of the fuel assembly are covered, as noted previously in this safety evaluation, the data for large pipe diameters, such as reactor coolant system pipes, are not extensive and use of the Chexal-Lellouche model will need justification. Assessment work indicates that the model tends to under-predict the void profile in the range of 12 to 17 MPa. In addition, the accuracy of the model in the range of 7.5 to 10 Mpa, which covers BWR ATWS conditions, has not been fully demonstrated. Results of analyses using the model in these ranges must be carefully reviewed.

The Chexal-Lellouche correlation cannot be used in situations where CCFL is important unless validation for appropriate geometry and expected flow conditions is provided.

Duke Position: [ ] This use is consistent with the Staff Position.

17. While FRIGG test comparisons have been presented for the dynamic slip option the issues concerning the Schrock-Grossman round tube data comparisons should be resolved before the dynamic slip option is approved. Plant comparisons using the option should also be required.

Staff Position: Assessment analyses (Reference 4), have shown that "the issues concerning the Schrock-Grossman round tube data comparisons" (actually the Bennett round tube data) are due to early prediction of CHF, which is nearly independent of the slip model used. Since the issue raised in the limitation is not related to the dynamic slip model, the limitation is considered to be resolved. The dynamic slip model is approved for use as given in condition (10).

Duke Position: Resolved per the Staff Position.

18. The nonequilibrium pressurizer model has no fluid boundary heat losses, cannot treat thermal stratification in the liquid region and assumes instantaneous spray effectiveness and a constant rainout velocity. A constant UA is used and flow detail within the component cannot be simulated. There will be a numerical drift in energy due to the inconsistency between the two-region and the mixture energy equations but it should be small. No comparisons were presented involving a full or empty pressurizer. Specific application of this model should justify the lack of fluid boundary heat transfer on a conservative basis.

Staff Position: The concern raised in this limitation of use is partially resolved in RETRAN-3D. Wall heat transfer can be included in the RETRAN-3D pressurizer model. Including wall heat transfer resolves this concern.

While the model does not directly account for thermal stratification, its effects can be included by use of normal nodes below the pressurizer volume. The user will have to justify the lack of thermal stratification or the use of normal nodes below the pressurizer should there be an indication that it would be important in the analysis.

The mixture and two-region energy equations are consistent for the implicit solution method where the mixture energy equation is used with the vapor-region energy equation. This eliminates inconsistency between the two-region and mixture energy equations and the concern regarding a potential drift in the region energies.

The staff notes that when a pressurizer fills or drains, a single region exists for which the normal pressure equation of state is used. Lack of numerical discontinuities in validation analyses of filling and draining pressurizers indicates that the model is functioning properly. It is the responsibility of the code user to justify any numerical discontinuity in the pressurizer during a filling or draining event.

The pressurizer model has options that require user-supplied parameters. Users must provide justification for these model parameters.

Duke Position: Duke is not proposing any changes in modeling the pressurizer with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA. The good modeling practices in the Staff Position are noted.

19. The non-mechanistic separator model assumes quasi-statics (time constant approximately few tenths of seconds) and uses GE BWR6 carryover/carryunder curves for default values. Use of default curves has to be justified for specific applications. As with the pressurizer a constant L/A is used. The treatment in the off normal flow quadrant is limited and those quadrants should be avoided. Attenuation of pressure waves at low flow/low quality conditions are not simulated well. Specific applications to BWR pressurization transients under those conditions should be justified.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not use the separator model.

20. The centrifugal pump head is divided equally between the two junctions of the pump volume. Bingham pump and Westinghouse pump data are used for the default single-phase homologous curves. The SEMISCALE MOD-1 pump and Westinghouse Canada data are for the degradation multiplier approach in the two-phase regime. Use of the default curves has to be justified for specific applications. Pump simulation should be restricted to single-phase conditions.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: As noted in the Staff Position, the basic model in RETRAN-3D is unchanged from RETRAN-02. Duke is not proposing any changes in modeling the reactor coolant pumps with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA, Section 2.2.6.2.

21. The jet pump model should be restricted to the forward flow quadrant as the treatment in the other quadrants is conceptually not well founded. Specific modellng of the pump in terms of volumes and junctions is at the user's discretion and should therefore be reviewed with the specific application.

Staff Position: Subsequent revisions of RETRAN-02 addressed this limitation. Since RETRAN-3D has the same model as RETRAN-02 MODOO3, and subsequent versions, their acceptance applies to RETRAN-3D.

Duke Position: Duke does not model BWR jet pumps.

22. The non-mechanistic turbine model assumes symmetrical reaction staging, maximum stage efficiency at design conditions, a constant UA and a pressure behavior dictated by a constant loss coefficient It should only be used for quasi-static conditions and in the normal operating quadrant.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not use the turbine model.

23. The subcooled void model is a nonmechanistic profile fit using a modification of EPRI recommendations for the bubble departure point It is used only for the void reactivity computation and has no direct effect on the thermal-hydraulics. Comparisons have only been presented for BWR situations. The model should be restricted to the conditions of the qualification data base. Sensitivity studies should be requested for specific applications. The profile blending algorithm used will be reviewed when submitted as part of the new manual (MODOO3) modifications.

Staff Position: The profile blending algorithm approved for RETRAN-02 MODOO3 is used in RETRAN-3D therefore this condition has been satisfied.

Duke Position: Resolved per the Staff Position.

24. The bubble rise model assumes a linear void profile, a constant rise velocity (but adjustable through the control system), a constant UA, thermodynamic equilibrium, and makes no attempt to mitigate layering effects. The bubble mass equation assumes zero junction slip which is contrary to the dynamic and algebraic slip model. The model has limited application and each application must be separately justified.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies. However, the layering effects encountered in RETRAN-02 can be eliminated using the RETRAN-3D stack model. This partially resolves the concern by resolving the layering limitation through use of the stack model.

Duke Position: As noted in the Staff Position, the basic model in RETRAN-3D is unchanged from RETRAN-02. Duke is not proposing any changes in using the bubble rise model with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA, Section 2.2.6.4. Duke does not currently stack bubble rise volumes, but if future modeling does, the stack model will be used.

25. The transport delay model should be restricted to situations with a dominant flow direction.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies. The appropriate application of the model is for one-dimensional flow. The user will have to justify use of this option in the absence of a dominant flow direction.

Duke Position: As noted in the Staff Position, the basic model in RETRAN-3D is unchanged from RETRAN-02. Duke is not proposing any changes in using the transport delay model with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA, Section 2.2.7.4. The limitation with applying this model without a dominant flow direction is well known and is avoided.

26. The stand-alone auxiliary DNBR model is very approximate and is limited to solving a one-dimensional steady-state simplified HEM energy equation. It should be restricted to indicating trends.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not use the auxiliary DNBR model.

27. Phase separation and heat addition cannot be treated simultaneously in the enthalpy transport model For heat addition with multidirectional, multifunction volumes the enthalpy transport model should not be used without further justification. Approval of this model will require submittal of the new manual (MODOO3) modifications.

Staff Position: A number of the simplifying assumptions in the RETRAN-02 enthalpy transport model have been eliminated in RETRAN-3D which now allows multiple inlet and outlet flows

and eliminates the simplifying assumptions related to mass distribution and pressure change effects. This condition has been adequately addressed.

Duke Position: Resolved per the Staff Position

28. The local conditions heat transfer model assumes saturated fluid conditions, one-dimensional heat conduction and a linear void profile. If the heat transfer is from a local condition volume to another fluid volume, that fluid volume should be restricted to a non-separated volume. There is no qualification work for this model and its use will therefore require further justification.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position:

29. The initializer does not absolutely eliminate all ill-posed data and could have differences with the algorithm used for transient calculations. A null transient computation is recommended. A heat transfer surface area adjustment is made and biases are added to feedwater inlet enthalpies in order to justify steady-state heat balances. These adjustments should be reviewed on a specific application basis.

Staff Position: The over-specified condition is identified by the RETRAN-3D steady-state input checking, resolving the concern regarding ill-posed data. The user must still run null transients to ensure that unwanted control or trip actions are not affecting the transient solution.

RETRAN-3D has available a low power steady-state steam generator initialization option that eliminates the heat conductor area change used in the RETRAN-02 initialization scheme. When this option is used, no adjustments are made to the heat transfer area and this specific concern is resolved. However, either the pressure or temperature is adjusted on the secondary side. These adjustments should be reviewed by the user on a specific application basis. The low power steady-state initialization option is approved for use.

Duke Position: Resolved per the Staff Position.

30. Justification of the extrapolation of FRIGG data or other data to secondary-side conditions for PWRs should be provided. Transient analysis of the secondary side must be substantiated. For any transients in which two-phase flow is encountered in the primary all the two-phase flow models must be justified.

Staff Position: The Chexal-Lellouche correlation is approved for use with PWR applications as stated in conditions (10) and (16). The user must justify choosing any other two-phase flow correlation.

Duke Position: [.]
Resolved per the Staff Position.

31. The pressurizer model requires model qualification work for the situations where the pressurizer either goes solid or completely empties.

Staff Position: The pressurizer model is approved for use with filling and draining events as given in condition (18).

Duke Position: Resolved per the Staff Position

32. Transients which involve three-dimensional space-time effects such as rod ejection transients would have to be justified on a conservative basis.

Staff Position: The 3-dimensional kinetics model, as noted in limitation 1 above, satisfies this limitation.

Duke Position: Resolved per the Staff Position. Duke is not using the three-dimensional model.

33. Transients from subcritical, such as those associated with reactivity anomalies should not be run.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke is not running any transients from subcritical.

34. Transients where boron injection is important, such as steamline break will require separate justification for the user-specified boron transport model.

Staff Position: The generalized transport model was added to RETRAN-3D to provide the capability to track materials such as boron. Specific application of the model to steam line break transients must be justified by the user. The model is approved for use as given in condition (3).

Duke Position: The generalized transport model is unchanged from RETRAN-02 relative to its use for modeling boron transport. The generalized transport model was approved in the SER for RETRAN-02 MOD005.0 dated November 1, 1991. Duke's use of this model for Oconee

emergency boron injection is documented in topical report DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology," Revision 1, Section 15.3.1.1.3. This topical report was approved by the NRC with SERs dated October 1, 1998 and May 25, 1999. No further NRC review is necessary.

35. For transients where mixing and cross flow are important, the use of various cross flow loss coefficients has to be justified on a conservative basis.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke models mixing and cross flow in the reactor vessel during transients and accidents in which loop asymmetry is important. Duke's modeling for Oconee is documented in topical report DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology," Revision 1, Section 15.2.1.1. This topical report was approved by the NRC with SERs dated October 1, 1998 and May 25, 1999. No further NRC review is necessary.

36. ATWS events will require additional submittals.

Staff Position: RETRAN-3D is approved for PWR ATWS analyses as given in condition (13).

Duke Position: Resolved per the Staff Position.

37. For PWR transients where the pressurizer goes solid or completely drains, the pressurizer behavior will require comparison against real plant or appropriate experimental data.

Staff Position: The pressurizer model is approved for use with filling and draining events as noted in the discussion of conditions (18) and (31).

Duke Position: Resolved per the Staff Position

38. PWR transients, such as steam generator tube rupture, should not be analyzed for two-phase conditions beyond the point where significant voiding occurs on the primary side.

Staff Position: The use of slip models for PWR applications is approved for use as given in conditions (16) and (30).

Duke Position: In the Oconee steam generator tube rupture UFSAR Chapter 15 analysis significant voiding does not occur on the primary side. However, significant voiding can occur on the primary side for steam line break events. Duke's UFSAR Chapter 15 steam line break modeling for Oconee is documented in topical report DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology," Revision 1, Chapter 15. This topical report was approved by the NRC with SERs dated October 1, 1998 and May 25, 1999. Duke's UFSAR Chapter 6 steam line break mass and energy release modeling for Oconee is documented in topical report DPC-NE-3003-PA, "Mass and Energy Release and Containment Response Methodology," Chapter 5. This topical report was approved by the NRC with SER dated March 15, 1995. Modeling of two-

[

]No further NRC review is necessary.

39. BWR transients were asymmetry leads to reverse jet pump flow such as the one recirculation pump trip, should be avoided.

Staff Position: As noted in the discussion of condition (21), this is resolved.

Duke Position: Duke does not model BWRs.

40. Organizations with NRC-approved RETRAN-02 methodologies can use the RETRAN-3D code in the RETRAN-02 mode without additional NRC approval, provided that none of the new RETRAN-3D models listed in the definition are used. Organizations with NRC-approved RETRAN-02 methodologies must obtain NRC approval prior to applying any of the new RETRAN-3D models listed above for UFSAR Chapter 15 licensing basis applications. Organizations without NRC-approved RETRAN-02 methodologies must obtain NRC approval for such methodologies or a specific application before applying the RETRAN-02 code or the RETRAN-3D code for UFSAR Chapter 15 licensing basis applications. Generic Letter 83-11 provides additional guidance in this area. Licensees who specifically reference RETRAN-02 in their technical specifications will have to request a Technical Specification change to use RETRAN-3D.

Duke Position: The submittal of DPC-NE-3000-P, Revision 3, includes the use of the [

]The SER on p. 33 also states that use of the "new control blocks added to improve functionality" requires NRC review and approval. The new control blocks in RETRAN-3D are the following:

ABS - Absolute value
F2D - Two-dimensional interpolation
RAT - Rate
STF - Second-order transfer function

None of these new control block models have yet been incorporated into any of the Duke RETRAN models used for licensing basis applications. However, use of some of these new control block models in the future is likely to enhance and simplify applications. Since all of these new control blocks consist of well-founded arithmetic and mathematical formulas, similar to the control blocks included in RETRAN-02, it is not understood why NRC approval prior to their use is necessary. Duke requests NRC approval to use the new RETRAN-3D control blocks for future applications consistent with their formulation.

41. RETRAN may be used for BWR ATWS subject to the following restrictions: The licensee must validate the chosen void model over the range of pressure, channel inlet flow, and inlet subcooling encountered during the transient that are outside the range of conditions for which assessment is available. Furthermore, the licensee should validate the choice of steam separator model and evaluate its use relative to steam separator performance data relevant to the conditions

present during the ATWS simulation. The licensee must also evaluate the uncertainties in the modeling. See Condition (16) and the Staff Position for related information.

Duke Position: Duke does not model BWRs.

42. The RETRAN-3D five-equation, or nonequilibrium, model uses flow regime maps and flow pattern dependent heat transfer and interfacial area models to simulate the heat and mass transfer processes between phases. A licensee wishing to apply the five-equation model will have to justify its use outside areas of operation where assessment has been documented. This may include either separate effects or integral systems assessment that cover the range of conditions encountered by the application of interest. An assessment of the uncertainties must also be provided. The model is approved subject to these conditions.

Duke Position: Duke does not use the five-equation model for licensing basis applications.

43. Assessment performed in support of use of RETRAN-3D must also address consistency between the RETRAN-3D calculations and any auxiliary calculations that are a part of the overall methodology, such as, departure from nucleate boiling or critical power ratio.

Duke Position: Duke uses results from RETRAN-3D analyses for input to other codes to perform core power distribution analyses, detailed core thermal-hydraulic analysis of the departure from nucleate boiling phenomenon, fuel rod and pellet thermal and mechanical behavior analyses, and containment thermal and structural response to high-energy line breaks. The details of these other methodologies have been submitted and approved by the NRC as appropriate. Any revisions to these methodologies, including any changes due to the use of RETRAN-3D in place of RETRAN-02, will be submitted for NRC review prior to their use for licensing basis applications.

44. The staff concludes that the lack of a detailed RETRAN-3D specific user guideline document mandates a statement on the user's experience and qualification with the code when analyses are submitted in support of licensing actions. This statement is expected to be consistent with the guidance of Generic Letter 83-11.

Duke Position: It is noted that Volume 3 of the EPRI RETRAN-3D documentation set has been enhanced subsequent to the NRC SER to include a significant amount of user guidelines regarding modeling option selection, in particular for the new RETRAN-3D models and options. Revision 3 to DPC-NE-3000-P fully describes Duke's use of the RETRAN-3D code for simulating the Oconee Nuclear Station with replacement steam generators. This revision is submitted for NRC review with the intent of maintaining the documentation of the Duke RETRAN methodology current, along with the main purpose of obtaining NRC review and approval for the transition from RETRAN-02 to RETRAN-3D for Oconee. This topical report revision extends Duke's response to Generic Letter 83-11. Duke's current level of RETRAN user experience is 15 engineers with a total of 144 years of experience with RETRAN-02 and RETRAN-03/-3D.

45. Assessment of the RETRAN-3D code for the models not explicitly approved in this safety evaluation will be the responsibility of the licensee or applicant. In addition, application of the

RETRAN-02 or RETRAN-3D codes for best estimate analysis of UFSAR Chapter 15 licensing basis events may require additional code and model assessment, and an evaluation of uncertainties to assure accurate prediction of best estimate response. This condition is based on the absence, in the best estimate analysis approach, of the conservative assumptions in traditional UFSAR Chapter 15 licensing basis analyses. For each use of RETRAN-3D in a licensing calculation, it will be necessary for a valid approach to assessment to be submitted, which is expected to include a PIRT for each use of the code and the appropriate assessment cases and their results. The scope of the PIRT and validation/assessment will be commensurate with the complexity of the application.

Duke Position: Duke has previously received NRC review and approval for application of the RETRAN-02 code to the licensing basis applications for non-LOCA transients and accidents for the Oconee Nuclear Station. The three RETRAN-related topical reports and associated NRC SERs supporting Oconee are:

> DPC-NE-3000-PA, Revision 2, "Thermal-Hydraulic Transient Analysis Methodology", December 2000. SERs are dated 11/15/91 (Revision 0), 8/8/94 (Revision 1), and 12/27/95 (Revision 2)
>
> DPC-NE-3003-PA, "Mass and Energy Release and Containment Response Methodology", November 1997. SER is dated 3/15/95
>
> DPC-NE-3005-PA, Revision 1, "UFSAR Chapter 15 Transient Analysis Methodology, August 1999. SERs are dated 10/1/98 (Revision 0) and 5/25/99 (Revision 1)

These topical reports have all being revised and submitted for NRC review to address the Oconee replacement steam generators and use of the RETRAN-3D code for Oconee non-LOCA transient and accident analyses. Based on the close similarity of the replacement and original steam generators, the transient thermal-hydraulic behavior will be very similar. The only significant difference will be for the main steam line break analysis, in which the flow restricting orifices in the replacement steam generators steam outlet nozzles will effectively reduce the maximum break size and the blowdown rate.

activity was presented and reviewed by the NRC during the review of earlier revisions of DPC-NE-3000. [ ] has been incorporated into the Duke version of the RETRAN-3D code as described in Revision 3 to DPC-NE-3000-P.

In summary, Duke has previously obtained NRC review for RETRAN-02 modeling of Oconee with the original steam generators. Substantial validation and assessments comparisons were associated with the previous revisions to DPC-NE-3000. The designs of the original and replacement steam generators are very similar, and the transient performance will be very similar except for the response to large steam line break accidents. Revision 3 describes the use of RETRAN-3D for modeling Oconee with replacement steam generators. [ ] Duke is not proposing to use this model for best-estimate licensing applications. The traditional conservative approach will continue to be used for licensing applications. A PIRT is not being submitted due to the previous NRC approval of the Duke RETRAN methodology topical reports, the limited scope of changes in the methodologies, the similarity of the designs of the new and replacement Oconee steam generators, the use of only one new RETRAN-3D model, and the assessment that has been performed to justify use of the one new model.

<u>References</u>

A-1 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

A-2 RETRAN-3D MOD003.1DKE, SDQA-30218-NGO, Duke Power, October 30, 2001