September 24, 2003

Mr. Ronald A. Jones
Vice President, Oconee Site
Duke Energy Corporation
P. O. Box 1439
Seneca, SC 29679

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 - SAFETY EVALUATION OF REVISIONS TO TOPICAL REPORTS DPC-NE-3000, -3003, AND -3005 (TAC NOS. MB5441, MB5442, AND MB5443)

Dear Mr. Jones:

By letter dated June 13, 2002, you submittal the following revisions to three topical reports: (1) DPC-NE-3000-P, Revision 3, “Thermal-Hydraulic Transient Analysis Methodology”; (2) DPC-NE-3003-P, Revision 1, “Mass and Energy Release and Containment Response Methodology”; and (3) DPC-NE-3005-PA, Revision 2, “UFSAR Chapter 15 Transient Analysis Methodology.” You asked for approval of three revisions to support the replacement of the steam generators at Oconee Nuclear Station, Units 1, 2, and 3. You provided additional information in your letters dated May 21, July 7, and July 28, 2003.

Enclosure 1 contains our Safety Evaluation (SE) of DPC-NE-3000-P, Revision 3, and DPC-NE3005-PA, Revision 2; and Enclosure 2 contains our SE of DPC-NE-3003-P, Revision 1.

Sincerely,

***/RA/***

Leonard N. Olshan, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosures: As stated

cc w/encls: See next page

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|---|---|---|---|---|---|
| NAME | LOlshan | CHawes | FAkstulewicz* | RDennig* | JNakoski |
| DATE | 9/24/2003 | 9/16/2003 | 8/14/03 | 9/3/03 | 9/24/03 |

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cc:

Ms. Lisa F. Vaughn
Duke Energy Corporation
Mail Code - PB05E
422 South Church Street
P.O. Box 1244
Charlotte, North Carolina 28201-1244

Anne W. Cottingham, Esquire
Winston and Strawn
1400 L Street, NW
Washington, DC 20005

Manager, LIS
NUS Corporation
2650 McCormick Drive, 3rd Floor
Clearwater, Florida 34619-1035

Senior Resident Inspector
U. S. Nuclear Regulatory Commission
7812B Rochester Highway
Seneca, South Carolina 29672

Mr. Henry Porter, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201-1708

Mr. Michael A. Schoppman
Framatome ANP
1911 North Ft. Myer Drive
Suite 705
Rosslyn, VA 22209

Mr. L. E. Nicholson
Compliance Manager
Duke Energy Corporation
Oconee Nuclear Site
7800 Rochester Highway
Seneca, South Carolina 29672

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of Justice
P. O. Box 629
Raleigh, North Carolina 27602

Mr. C. Jeffrey Thomas
Manager - Nuclear Regulatory Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

Mr. Richard M. Fry, Director
Division of Radiation Protection
North Carolina Department of Environment, Health, and Natural Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

Mr. Peter R. Harden, IV
VP-Customer Relations and Sales
Westinghouse Electric Company
6000 Fairview Road
12th Floor
Charlotte, North Carolina 28210

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR

REGULATION OF REVISION 3 TO DPC-NE-3000-P "THERMAL-HYDRAULIC

TRANSIENT ANALYSIS METHODOLOGY" AND REVISION 2 TO DPC-NE-3005-P,

"UFSAR CHAPTER 15 TRANSIENT ANALYSIS METHODOLOGY"

## 1.0 INTRODUCTION

The Duke Energy Corporation (the licensee) is making preparations to replace the steam generators at the Oconee Nuclear Station. The replacement once-through steam generators (ROTSGs) are essentially of the same once-through design as the original once-through steam generators (OTSGs), but there are a number of small differences that call for revisions to the previously-approved topical reports. This safety evaluation involves the revisions to two topical reports namely: DPC-NE-3000-P "Thermal-Hydraulic Transient Analysis Methodology" and DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology." The licensee has reanalyzed several design basis transients and accidents in the Oconee Updated Final Safety Analysis Report (UFSAR). The licensee used the RETRAN-3D thermal/hydraulic computer code rather than the RETRAN-02 code that is the current approved analytical method to perform the transient reanalyses. Other revisions to the topical reports include editorial and minor technical changes. The licensee requested NRC review by letter dated June 13, 2002 (Ref. 1), and the licensee provided additional information in a letter dated May 21, 2003 (Ref. 2).

## 2.0 REGULATORY EVALUATION

The NRC staff approved the generic use of RETRAN-3D by licensees as discussed in Ref. 3. The licensee plans to utilize RETRAN-3D in a manner that causes the code to essentially default to be the same as RETRAN-02. In Ref. 3, the NRC staff stated that 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 are used.

The licensee has selected to use certain of the new RETRAN-3D models. In addition, the licensee requested the RETRAN code developers to make several Oconee specific changes in RETRAN-3D. These changes include options to the critical flow model, to the vertical steam/water separation model and forced convection heat transfer.

As part of its approval of RETRAN-3D, the NRC staff included 45 conditions that users of RETRAN-3D must address before using the code. The licensee provided responses to each of these conditions. Since the licensee does not utilize the majority of the new RETRAN-3D features, most of these conditions do not apply. Those instances, where the licensee has deviated from the RETRAN-02 mode either as a result of using RETRAN-3D features or as a

result of code modifications, are evaluated in the following section. The NRC staff will approve use of the Oconee-specific RETRAN-3D model if the licensee demonstrates that the options added provide conservative results, or more realistic or accurate modeling of the plant response; and the specific limitations and conditions imposed generically on the RETRAN-3D code have been satisfied.

## 3.0 TECHNICAL EVALUATION

A significant difference between the ROTSGs and the OTSGs is the addition of a flow restriction in the exit nozzles. Flow restrictions are common in pressurized water reactors of a more recent design than Oconee. Flow restrictions limit the rate of steam release in the event of a main steam line break and limit the rate of reactor system cooldown. In addition, the ROTSGs provide a small increase in the number of tubes and available heat transfer area compared to the OTSGs. There is slightly more water in the ROTSGs and the steam generator tube material has been modified. Because of the physical changes, the licensee believed it necessary to reevaluate certain of the transients and accidents in the plant UFSAR.

The licensee made the required reevaluations using RETRAN-3D. RETRAN-3D has the benefit of being a newer computer code than RETRAN-02 and has incorporated error corrections by the code developers based on the experience of the users. The licensee has utilized RETRAN-3D in a mode that essentially defaults to RETRAN-02 with certain exceptions:

- With RETRAN-3D the steam generators can now be modeled using separate velocities for the steam and water flowing up a tube bundle. The model was benchmarked against the steam generator design codes. The licensee requested code modifications to allow the relative velocities between the steam and water phases to be adjusted to produce the appropriate steam generator inventory. The NRC staff views this modification as an improvement in code accuracy and modeling capability. It is, therefore, acceptable.

- The licensee extended the heat transfer capability of RETRAN-3D to allow for condensation heat transfer when the surface temperature of a conductor is lower than that of steam in an adjacent channel. This modification is similar to one made in RETRAN-02 for the licensee by the code developers. The modification extends the accuracy of the code and is acceptable to the NRC staff.

- The licensee added a user option to calculate critical flow for a main steam line break assuming an inlet enthalpy corresponding to that of pure steam or two-phase conditions as predicted by the code from the level swell in the affected steam generator. Use of this model permits the licensee to conservatively perform main steam line break analyses assuming only steam exits the steam generator and is therefore acceptable to the NRC staff. The assumption of pure steam flow is used by the licensee for predicting mass and energy release to the containment. The licensee will retain the option of performing best estimate main steam line breaks in which the code calculates liquid entrainment.

- For analyses of the reactor system cooldown following a main steam line break, the licensee will utilize an enhanced steam separation velocity in the affected steam generator. Reactor system cooldown is evaluated to investigate the possibility of return to criticality in the core and departure from nucleate boiling (DNB). The use of an

enhanced steam velocity in the ROTSG will effectively eliminate liquid removal from the steam generator unless the steam generator completely fills with water from continued feedwater flow. Assumptions which cause water to remain in the steam generator and only steam to be removed are conservative for analysis of reactor system cooling. There is no steam separation equipment in the Oconee steam generators and the assumption of enhanced steam separation is made to ensure conservative results. These assumptions provide for a conservative prediction of reactor cooldown following a main steam line break and are, therefore, acceptable to the NRC staff.

- The licensee added options to the forced convection model so as not to over predict heat transfer from the steam generator structural metal to the exiting fluid during a main steam line break. This modification provides for more accurate predictions of steam generator stresses following a main steam line break and is, therefore, acceptable to the NRC staff.

The licensee provided the NRC staff with the updates to RETRAN-3D for each of these modifications. The NRC staff reviewed the source coding and confirmed that in each case the modification was performed by the code developer and independently verified. With the exception of these modifications, the licensee will use RETRAN-3D in the RETRAN-02 mode. The NRC staff reviewed the responses by the licensee to all of the generic conditions required of users of RETRAN-3D. The NRC staff agrees with the licensee’s assessment that in each case the condition either does not apply because that feature of the code is not utilized by the licensee or the proposed use of the code by the licensee meets the NRC requirement.

Topical report DPC-NE-3005-P describes the details of how RETRAN-3D is utilized to analyze reactor system transients and accidents. The plant noding description is essentially unchanged for analyses with the new steam generators. One difference in methodology is the use of MCNP Monte-Carlo transport code (Ref. 4) to evaluate neutron attenuation to the excore detectors. The neutron source for input into MCNP is evaluated using SAS2H/ORIGEN-S modules of the SCALE code system (Ref. 5). This methodology is commonly used in the nuclear industry and will provide increased accuracy over the methodology currently in use. The effect of downcomer temperature on neutron attenuation is important in evaluating the differences between measured and actual reactor power for the high power reactor trip determination. This determination is important to the analyses of control rod misalignment events and small steam line breaks.

RETRAN-3D is programmed to incorporate the ANS-5.1 decay heat standard of 1979. The RETRAN-3D code does not include the contribution from neutron capture within stable fission products that is part of the standard. The licensee accounts for this omission by inputting a decay heat correction multiplier table to the code input. The table includes the addition of two standard deviations to the decay power. This approach provides a high confidence that the actual decay heat will be bounded by the RETRAN-3D calculation and is, therefore, acceptable to the NRC staff. The NRC staff has accepted a similar approach for use with RETRAN-02. This upper bound of decay heat is applied for applications when it is conservative for decay heat to be high. For applications when it is conservative to have low values of decay heat, such as to evaluate overcooling following a main steam line break, the licensee uses conservatively low values of the decay heat multiplier.

To demonstrate the effect of using RETRAN-3D for analysis when the new steam generators are installed, the licensee performed a limited number of plant transient and accident analyses. These results will be added as a revision to the UFSAR. The specific revised analyses were for large and small main steam line breaks and turbine trip.

A turbine trip causes a sudden cessation of main steam flow. The effect is to cause primary and secondary system pressures to increase. The main steam safety valves open and the reactor trips on high primary system pressure. With the ROTSGs installed and using the RETRAN-3D computer code, the reactor system pressure was calculated to reach 2595 psig. This pressure is within 110 percent of design pressure and, therefore, meets the acceptance criteria of the NRC staff's Standard Review Plan. The maximum reactor system pressure calculated for a turbine trip using RETRAN-02 and the OTSGs was 2611.8 psig. The closeness of the results shows there is little change between the older and newer versions of RETRAN for overpressure events. The change in steam generators would not be expected to significantly affect the consequences for events of this type.

The rupture of a main steam line causes a rapid increase in reactor system heat removal and a decrease in reactor system temperature and pressure. The licensee analyzed two cases of large main steam line breaks at full power. A case for which offsite power remains available was analyzed to determine the maximum overcooling. A case for which offsite power was lost was analyzed to determine if DNB would occur on the fuel pins. At Oconee, the steam generators contain the maximum water mass at full power so that a steam line break from full power will be the most severe condition for overcooling.

The Oconee plants are equipped with an automatic feedwater isolation system to prevent continued feedwater flow into the steam generators in the event of a main steam line break. The system is not fully safety-related so the licensee does not take credit for operation of this system. For the case with offsite power available, the feedwater pumps are assumed to continue to operate filling both steam generators. The licensee analyzed the consequences of a main steam line break with continued feedwater pump operation in the previous revision to DC-NE-3005-P using RETRAN-02. These analyses were for the OTSGs.

With offsite power available, the reactor coolant pumps would remain in operation and maximize the rate of cooldown. If the reactor is sufficiently cooled, reactivity feedback from the moderator might cause the core to return to criticality with the control rods inserted. Return to criticality would be of particular concern in the event one control assembly failed to insert in the core causing power to peak at that location. In previous analyses using RETRAN-02 with the OTSGs, the reactor core was calculated to return to criticality if the most reactive control rod did not enter the core. The DNB limits for the core were shown to not be exceeded for this condition. In the analysis with the ROTSG, the core does not become critical again. This is because the flow restriction in the steam generator nozzles delays the cooldown and provides more time for boric acid to be pumped into the reactor core from the core flood tanks and the high pressure injection system. The licensee verified that the core will remain subcritical by inputting the thermal/hydraulic conditions calculated for the core into the three dimensional SIMULATE-3P neutronics code. The SIMULATE-3P results confirmed that the core will remain subcritical.

As part of the NRC staff review of the previous revision to DC-NE-3005-P (Ref. 6) the NRC staff preformed audit calculations of postulated main steam breaks using RELAP5 and compared

the results of these audit calculations to the licensee's calculations using RETRAN-02. The NRC staff uses audit calculations as an aid in understanding and evaluating the sequences and phenomena in postulated reactor accidents. Conclusions on the acceptability or unacceptability of license applications are based on licensee calculations using approved methodology and not on the results of the NRC staff audit.

In previous analysis using RETRAN-02, the licensee predicted that the maximum overcooling following a main steam line break would occur if the feedwater control system continued to function. This was in conflict with the NRC staff audit using RELAP5 that indicated that maximum overcooling would occur if the feedwater control system failed. The new analysis using RETRAN-3D agrees with the NRC staff's prediction that failure of feedwater control system is the worst case. In the previous analysis using RETRAN-02, the reactor coolant pumps in the unaffected coolant loop became unstable and had to be tripped to keep the code from failing. In the RETRAN-3D analyses the pumps do not become unstable and continue to run. This result is also similar to the NRC staff's audit.

For the case of a large steam line break with loss of offsite power, the concern is loss of the required core DNB margin as a result of the reactor coolant pump coastdown and the decrease in reactor system pressure. The licensee uses maximum values of decay heat for this calculation to conservatively calculate the DNB margin. The thermal/hydraulic conditions predicted for the core by RETRAN-3D are input into the VIPRE code to calculate the minimum margin to DNB. The minimum DNB margin was found to be increased from the previous analysis primarily as a result of the action of the flow restriction in the steam generator nozzles. The licensee uses critical heat flux correlations that have been approved by the NRC staff to evaluate the DNB margin.

The licensee analyzed a spectrum of small steam line breaks to determine the effect of the ROTSGs. The limiting cases for small steam line breaks do not result in a reactor trip due to the reduction in reactor vessel downcomer temperature affecting the excore neutron detectors. The resulting margin to DNB was found to be increased over the previous analysis with the OTSGs.

The licensee has stated that the revised analyses for turbine trip and large and small steam line breaks will be incorporated into the Oconee UFSAR.

In addition to the technical revisions to topical reports DPC-NE-3000-P and DPD-NE-3005, the licensee has proposed minor editorial changes. The NRC staff has reviewed these changes and finds all of them to be acceptable.

## 4.0 CONCLUSIONS

On the basis of its review of Revision 3 to DPC-NE-3000-P and Revision 2 to DPC-NE-3005-P, including supplemental information provided by the licensee, the NRC staff concludes (1) that the modifications to RETRAN-3D will result in conservative results, and (2) that the licensee has demonstrated that the limitations and conclusions of the generic safety evaluation for RETRAN-3D are met. Therefore, the NRC staff concludes that RETRAN-3D as modified is acceptable for use at Oconee. In addition, the licensee has adequately applied RETRAN-3D to the safety analyses for Oconee in accordance with the requirements of the NRC staff's safety evaluation for RETRAN-3D (Ref. 3) and has made other revisions to the topical reports that are

acceptable to the NRC staff. The methodology in Revision 3 to DPC-NE-3000-P and Revision 2 to DPC-NE-3005-P is therefore approved and found acceptable for performing UFSAR Chapter 15 transient and accident analyses at Oconee. Use of this methodology for applications other than described in the topical reports will require additional NRC staff review and approval. The plant analyses contained in the topical reports are typical of those that will be incorporated into the UFSAR. For subsequent core reloads or other plant modifications, the licensee should verify that the analyses in the topical reports and UFSAR bound the results that would be obtained for the new plant condition or should perform new analyses that are conservative for that purpose.

5.0 REFERENCES

1. Letter from M. S. Tuckman, Duke Energy Corporation, to Chief, Information Management Branch, NRC, “Revisions to Topical Reports DNC-NE-3000, 3003, and 3005 in Support of Steam Generator Replacement, June 13, 2002.

2. Letter from K. S. Canady, Duke Energy Corporation, to Chief, Information Management Branch, NRC, “Revisions to Topical Reports DPC-NE-3000 and 3005 in Support of Steam Generator Replacement Response to NRC Staff Requests for Additional Information, May 21, 2003.

3. Letter from Stuart A. Richards, NRC to Gary 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 21, 2001.

4. Judith F. Briesmeister, Ed., “MCNP-A General Monte Carlo N-Particle Transport Code,” Los Alamos National Laboratory Report, LA-13709-M, March 2000.

5. “SCALE Version 4.4, A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation,” Oak Ridge National Laboratory, ORNL/NUREG/CSD-2/R6, May 2000.

6. Letter from David E. LaBarge, NRC to W. R. McCollum, Jr., Duke Energy Company, Oconee Nuclear Station, Units 1, 2, and 3 RE: Safety Evaluation for Revision 1 to Topical Report DPC-NE-3005-P, “UFSAR Chapter 15 Transient and Accident Analysis Methodology,” May 25, 1999.

Principal Contributor: W. Jensen, SRXB

Date: September 24, 2003

SAFETY EVALUATION BY OFFICE OF NUCLEAR REACTOR REGULATION

OF DUKE ENERGY CORPORATION TOPICAL REPORT DPC-NE-3003-P, REVISION 1

DOCKET NOS. 50-269, 50-270, AND 50-287

## 1.0 INTRODUCTION

An August 11, 1993, letter from Duke Energy Corporation to the NRC requested review and approval of Duke Energy Corporation topical report DPC-NE-3003-P (Revision 0). The report describes methods of calculating the response of the Oconee Nuclear Station reactor building to a postulated design basis loss-of-coolant accident (LOCA) and a postulated design basis main steam line break (MSLB) accident. It includes the calculation of the mass and energy injected into the containment as a result of these events. The staff issued a March 15, 1995, safety evaluation (SE) approving this report for containment licensing calculations for the Oconee Nuclear Station.

The licensee revised this topical report in November 1997 and included Supplements 1 and 2. Supplement 1 is an MSLB analysis crediting the MSLB detection and feedwater isolation system. The addition of the MSLB detection and feedwater isolation system to the Oconee units and the accompanying analysis were approved by the NRC in a letter dated December 7, 1998. Supplement 1 replaced Chapters 5 and 6 of Revision 0. Supplement 2 is a comparison of the FATHOMS[1] containment analysis computer code with data from CVTR[2].

A June 13, 2002, letter from Duke Energy Corporation requested review and approval of DPC-NE-3003-P Revision 1 (Revision 1). The licensee provided supplemental information in letters dated July 7 and July 28, 2003. These supplements provide additional information and also discuss two additional changes to Revision 1 that were not included in the original Revision 1 report. Revision 1 incorporates changes to the methods described in the original topical report and Supplement 1, including a description of the proposed use of the GOTHIC 7.0 computer code[3] to perform containment thermal hydraulic calculations as well as the

[1]FATHOMS/Containment Analysis Package User's Manual, Version 2.4, Numerical Applications, Inc., April 20, 1989

[2]"Simulated DBA Tests of the Carolina Virginia Tube Reactor Containment - Final Report," R.C. Schmitt, G.E. Bingham, J.A. Norberg, Idaho Nuclear Corp., December 1970

[3]GOTHIC Containment Analysis Package Version 7.0, NAI-8907-02, Revision 13, July 2001

FATHOMS code and the replacement of RETRAN-02[4] with RETRAN-3D[5] for MSLB mass and energy release calculations. GOTHIC 7.0 is a general purpose thermal-hydraulics computer code developed by Numerical Applications, Incorporated for the Electric Power Research Institute (EPRI). RETRAN-3D was developed by Computer Simulation & Analysis, Inc. for EPRI. Its use for licensing calculations was approved by the NRC (with conditions and limitations) by a safety evaluation report (SER) dated January 25, 2001. Appendix A to Revision 1 discusses the application of RETRAN-3D to model the Oconee replacement steam generators. Appendix B to Revision 1 discusses compliance of this modeling with the conditions of the NRC SER approving RETRAN-3D. (The use of RETRAN-3D has been approved in Enclosure 1 that proceeds this SE.)

The computer codes discussed in Revision 1 are:

> RELAP5/MOD2-B&W[6] is used to calculate the mass and energy released from the reactor coolant system (RCS) following both large and small break LOCAs.
>
> The BFLOW code is used for long-term liquid and steam mass flow rates out of a large cold leg break in the reactor coolant system. Its description is provided in Revision 1.
>
> The RETRAN-3D code is used to calculate the mass and energy release for a main steam line break. This code was reviewed and approved by the NRC[7]. Revision 0 utilized RETRAN-02. Attachment 3 to the licensee's June 13, 2002, letter, Item 9, discusses some differences between RETRAN-02 and RETRAN-3D. In addition, Appendix B of the June 13, 2002, letter evaluates the use of RETRAN-3D for this application in terms of conditions and limitations of the NRC SER on RETRAN-3D.
>
> The FATHOMS/DUKE-RS[8] code is used to calculate the reactor building response to high energy line breaks.
>
> The GOTHIC 7.0 computer code is also used to calculate the reactor building response to high energy line breaks. Attachment 1 to the licensee's July 7, 2003, letter describes the important differences between FATHOMS/DUKE-RS and GOTHIC 7.0. Appendix C to Revision 1 provides a summary description of the GOTHIC 7.0 code.

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[4] RETRAN-02 A Program for Transient Thermal Hydraulic analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCMA Revision 4 EPRI November 1988

[5] 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

[6] B&W-10164P, Revision 1, "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water reactor LOCA and Non-LOCA Transient Analysis," Babcock and Wilcox, October 1988

[7] Letter from S.A. Richards (USNRC) to G.L. Vine (Electric Power Research Institute-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.

[8] CAP - Containment Analysis Package (FATHOMS 2.4), Numerical Applications, Inc., October 10, 1989

Revision 1 discusses containment analysis using the FATHOMS code. However, it is Duke Energy Corporation's intention to have the option to use GOTHIC 7.0 for these calculations.

Since Revision 0 has previously been approved by the NRC, this review will concentrate on the differences between Revision 0 and Revision 1. These changes are listed and evaluated in the Technical Evaluation section of this SE. Changes to the topical report that are editorial or descriptive are not discussed. The NRC staff agrees that the changes designated editorial are indeed editorial. In addition, the two changes to Revision 1 that were included in the July 7, 2003, letter but not discussed in the topical are discussed after the discussion of the items in the June 13, 2002, topical report submittal.

## 2.0 REGULATORY ANALYSIS

Containment pressure analyses are required for Oconee Nuclear Station as part of the design basis evaluation (DBE). All three Oconee units were licensed for construction prior to May 21, 1971. Therefore, the General Design Criteria of 10 CFR Part 50, Appendix A are not applicable to the Oconee Nuclear Station. Section 3.0, "Design of Structures, Components, Equipment and Systems," of the Oconee Updated Final Safety Analysis Report (UFSAR) lists the principal design criteria "developed in consideration of the seventy General Design Criteria proposed by the AEC (Atomic Energy Commission) in the Federal Register of July 11, 1967." Of these, the following are relevant to this SE:

> Criterion 10: *Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity, and together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.*
>
> Criterion 15: *Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.*
>
> Criterion 49: *The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design basis leakage rate, the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for the effects from metal water or other chemical reactions that could occur as a consequence of failure of the Emergency Core Cooling Systems.*

The licensee utilized guidance from two sources in establishing the analytical methods and assumptions for mass and energy release calculations and containment calculations. The first source is the NRC Standard Review Plan Sections 6.2.1, "Containment Functional Design," 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," and 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures."

The second source of guidance is the American National Standards Institute Standard ANSI/ANS-56.4-1983, "Pressure and Temperature Transient Analysis for Light Water Reactor Containments." The use of this ANSI/ANS standard is not required by the NRC. However, the ANSI/ANS standard provides guidance in making conservative assumptions for safety

calculations. Revision 0 used this guidance. Revision 1 also uses this guidance. In addition to this guidance, Revision 0 and Revision 1 both use sensitivity calculations to ensure that the input assumptions are conservative.

## 3.0 TECHNICAL EVALUATION

Revision 1 describes 76 changes to Revision 0. These changes are evaluated in this section. Some changes are editorial and are not discussed. Some of the 76 changes are identical but found in different sections of the report. Each change is numbered consistent with the item numbers in Revision 1. The section of Revision 1 for each change is identified.

In addition, the two changes to Revision 1 discussed in the July 7, 2003, letter are evaluated.

### Cover page and frontal pages

1. and 2. Editorial

### Chapter 1 Introduction

3., 4., 5. Editorial

### Chapter 2 RELAP5/MOD2-Babcock & Wilcox (B&W) - Primary System Mass and Energy Release

6. Section 2.1.2 Main feedwater piping

   The RELAP5 nodalization for the main feedwater piping is revised to include the main feedwater piping between the last check valve and the steam generator. The licensee states that this enables modeling flashing of the main feedwater in the piping if the steam generator pressure decreases low enough for the flashing to occur. If flashing occurs, additional hot water will be expelled into the steam generator with the potential to increase primary-to-secondary heat transfer. Revision 1 states that calculations show this to be a minor effect. It is added for completeness. Since it makes the model more accurate by modeling a real effect, this change is acceptable.

7. Section 2.1.2 Replacement Once-Through Steam Generator (ROTSG) Modeling

   Revision 1 states that the ROTSGs are modeled with the same noding as the original steam generators. The differences between the current and the replacement steam generators are described in Revision 1. In general, the licensee characterizes the replacement of the steam generators as "like-for-like." The input data for the ROTSGs has been recalculated to reflect the differences based on data supplied by the manufacturer, B&W Canada, Inc. Since the geometric configuration is very similar and the other input changes are minor, the NRC staff finds the proposed use of the same RELAP5 noding for the ROTSGs to be acceptable.

8. Editorial

9. New Section 2.5 RETRAN-3D code description

This section provides a brief discussion of RETRAN-3D. RETRAN-02 was used previously to model the mass and energy release from the MSLB. This was described in Revision 0. Appendix A of Revision 1, "Methodology Revision for Oconee Replacement Steam Generators," provides a description of the application of RETRAN-3D to steam line break mass and energy release calculations. Appendix B of Revision 1, "Evaluation of RETRAN-3D SER Conditions and Limitations for the Oconee RETRAN model with ROTSGs" discusses the compliance of the RETRAN-3D ROTSG modeling with the conditions of the January 25, 2001, NRC SER approving RETRAN-3D.

Modeling of Chapter 15 events with RETRAN-3D, including the MSLB mass and energy input for containment calculations, is evaluated in the SE contained in Enclosure 1 that proceeds this SE. The NRC staff considers this SE adequate to cover the types of analyses discussed in Revision 1.

10. A new Section 2.6 "GOTHIC 7.0 Code Description" is added to Revision 1.

Appendix C of Revision 1 provides a brief description of GOTHIC 7.0.

See the evaluation of Chapter 6 revisions later in this SE for an evaluation of the use of GOTHIC 7.0 for Oconee Nuclear Station containment thermal hydraulic analysis.

11. Editorial

<u>Chapter 3 Large Break LOCA Mass and Energy Release Analyses</u>

12. Section 3.1: The termination of the RELAP5 analysis is changed from 30 minutes to the end of the borated water storage tank (BWST) injection phase.

Revision 1 states that with further experience in applying the methods of Revision 0 it has become apparent that the appropriate time for terminating the RELAP5 analysis is at the transition from the BWST injection phase (that is, the time when the BWST has reached the level at which the emergency core cooling system (ECCS) pumps and reactor building spray (RBS) pumps switch suction to the emergency sump), not necessarily at 30 minutes, which was the criterion used in Revision 0.

At the transition from the BWST injection to recirculation, the water in the reactor vessel is in a steady state condition and the transition to the BFLOW code or the RELAP5 code (see Item 13), which calculates mass and energy release after blowdown, is more appropriately made. The NRC staff finds this change to be consistent with the accident scenario and it is therefore acceptable.

13. Section 3.1: Clarify that the BFLOW code is only used for cold leg break LOCA analysis and describe what is used for hot break analysis.

> Revision 0 stated: "The BFLOW code is used to calculate the mass and energy release for the remainder of the [LOCA] analysis."
>
> However, the BFLOW code was only intended for calculating the mass and energy release after blowdown for cold leg breaks. Revision 1 is revised to make this clear. RELAP5 calculates the post-blowdown mass and energy release for hot leg breaks. This change reflects current practice. RELAP5 is capable and acceptable for doing this type of calculation and the change is therefore acceptable.

14. Section 3.2: The initial pressurizer level is updated to be consistent with the technical specifications.

> The licensee proposes to use a value for pressurizer level that is consistent with the high level value of the technical specifications, plus instrument uncertainty. This is conservative for mass and energy release calculations and is, therefore, acceptable.

15. Section 3.2: Revision 1 states that a conservative high initial mass is assumed for the ROTSGs. The values are not specified.

> A conservatively high steam generator mass maximizes the energy transferred from the secondary to the primary during the LOCA mass and energy release and is therefore a conservative assumption. It is acceptable to not specify the numerical value of the steam generator operating level since this is an input value. The specification that the initial mass will be conservatively high is the significant condition for this topical report. Therefore, the licensee's change is acceptable.

16. Section 3.2: The temperature of the water in the core flood tanks is increased from 120 °F to 130 °F. Since the higher temperature is conservative for mass and energy release calculations, this is acceptable.

17. Section 3.2: Rather than specifying the RCS flow to be a low value, Revision 1 will specify that a sensitivity analysis must be done to determine whether low RCS flow or high RCS flow is more conservative with respect to peak containment pressure.

> The NRC staff determined this approach is acceptable since the conservative RCS flow condition will be used.

18. Editorial

19.  Section 3.3.1.1:  Bounding high values for fuel stored energy will be used.  In addition, the licensee's July 7, 2003, letter states that the gap conductance is held constant at its initial value.

> The NRC staff concludes that this is a bounding assumption since the gap conductance would actually decrease as the blowdown progresses and is, therefore, acceptable.

20.   Section 3.3.1.1:  The model of the main feedwater lines is revised to reflect the fact that the main feedwater system is realigned to feed through the upper auxilliary feedwater header following the trip of all four reactor coolant pumps.  The licensee states that this is to enhance natural circulation heat transfer.  This results in less feedwater flow following realignment.

> Since this modeling change reflects the design of the plant, the NRC staff finds this change to be acceptable.

21., 22., 23.  Editorial

24.  Section 3.3.2.1 of Revision 0 calculated the amount of metal-water reaction that takes place during the short term period of a postulated LOCA.  Revision 1 assumes a constant amount of 1 percent, which is the limit allowable by 10 CFR 50.46 for core-wide reaction of coolant with zirconium.

> The NRC staff determined that since 1-percent metal-water reaction is the maximum allowed by regulation, this is conservative and acceptable.

25.  Section 3.3.2.2:  The modeling of the low pressure injection (LPI) flow is revised to be consistent with the current plant design.  Following loss of 4160 V switchgear, flow from two high pressure injection (HPI) pumps and one LPI pump will be available for injection into the reactor vessel.

Previously a constant value for LPI flow was assumed.  The LPI flow is now modeled as a function of the RCS backpressure until it is throttled by the operator.

> Modeling the LPI flow as a function of RCS backpressure is more accurate than assuming a constant value.  Throttling the flow is consistent with the Oconee emergency operating procedures.  Therefore, the NRC staff concludes this change is acceptable.

26.  Section 3.3.2.2 of Revision 1 revises the reactor coolant pump two-phase head degradation model.

> The specific model is proprietary but it has been previously reviewed and approved by the NRC and is acceptable for this application.

27. Section 3.3.3.1: The calculation of the long term LOCA fission heat uses the same RELAP5 modeling used for the short term analyses.

> The NRC staff determined that this is an improvement in the long term analysis. Revision 0 uses fission heat values calculated by a separate computer code. In Revision 1 these calculations are done directly in the RELAP5 code that calculates the LOCA response. This also better reflects the initial and boundary conditions and the transient behavior. Thus, this is an improvement in modeling and is acceptable to the NRC staff.

28. Section 3.3.3.2: The long term large break LOCA containment analysis considers both cold leg pump discharge breaks and hot leg breaks. Revision 0 considered only a double ended guillotine break located at the A1 cold leg pump discharge.

> The NRC staff determined that this is an improvement in determining the limiting LOCA and is, therefore, acceptable.

29. Section 3.3.3.2: The assumed location and the size of the hot leg break for long term containment response to a LOCA is discussed. For the limiting containment environmental qualification response, the limiting break location is a double-ended guillotine break at the reactor vessel outlet nozzle. For the maximum containment sump water temperature analysis, the limiting break is a double-ended guillotine break located at the reactor vessel outlet nozzle.

> Since these findings are the result of calculations examining possible locations for the most conservative results, the NRC staff finds this selection of breaks acceptable.

30. Section 3.3.3.2 of Revision 1 discusses the timing of the depletion of the BWST and the switch to pump recirculation. Depending on the objective of the analysis (e.g., peak pressure, sump temperature), high or low ECCS flow rates are chosen.

> Since these findings are based on calculations examining possible break locations for the most conservative results, the NRC staff finds this selection of ECCS flow rates for BWST depletion to be acceptable.

31. Section 3.3.3.2: In Revision 0 the LPI pump flow was assumed constant. In Revision 1 the LPI flow rate is a function of RCS backpressure until the flow is throttled in accordance with the emergency operating procedures. In addition, in Revision 0 there is a discrepancy between the RBS flowrate assumed for the calculation of the rate of BWST depletion (the instrument uncertainty in flow rate is added to the nominal flow rate value) and the calculation of the RBS system flow rate in FATHOMS (the instrument uncertainty in flow rate is subtracted from the nominal flow rate value) in order to be conservative in each case. In Revision 1 these flow rates are the same.

> Revision 1 states that the changes to these analyses are necessary since plant procedures and station design have changed. Although the flow rates have been revised, Revision 1 states that conservative values are still used. Since the analysis is revised to reflect changes to plant procedures and design, and the

resulting flow rates used in the accident analyses are still conservative, these changes are acceptable to the NRC staff.

32. Section 3.3.3.4: For the long term LOCA containment analysis, the termination of the RELAP5 analysis is changed from 30 minutes to the end of the BWST injection phase.

See the evaluation of Item 12.

33. Editorial

Chapter 4 Small Break LOCA Mass and Energy Release

34. Section 4.1: Revision 0 described the use of a simple containment model in RELAP5 for the determination of containment conditions in small break LOCA mass and energy calculations. This has been revised in Revision 1. RELAP5 small break LOCA mass and energy calculations now use containment conditions from a FATHOMS analysis rather than a containment model in RELAP5.

There are two FATHOMS results that do impact the RELAP5 calculation. These are the RBS actuation time (the RBS actuates on containment high pressure) that affects the timing of the switch to recirculation from the containment sump and the containment sump water temperature. Iterations are performed between FATHOMS and RELAP5 until the results converge or conservative agreement is achieved between the FATHOMS and RELAP5. The licensee provided Table 9-1 in the July 7, 2003, letter that demonstrates good agreement between the sump temperature at the start of recirculation and the peak sump temperature for both FATHOMS and RELAP5. The NRC staff noted there is good agreement.

The licensee’s use of FATHOMS to determine the RELAP5 containment boundary conditions is acceptable to the NRC staff since FATHOMS is intended for calculations of this type (sump temperature and containment pressure).

35. Section 4.1: 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 in Revision 1. The stored energy in the secondary system fluid and structural metal is now included. FATHOMS now includes all heat sources. The modeling is done in FATHOMS with heater components. The licensee states that this is an improvement and a simplification.

The NRC staff concludes that inclusion of more heat sources is an improvement. Therefore, the NRC staff finds the change to be acceptable.

36. Section 4.3.1: Both Revision 0 and Revision 1 use the RELAP5 point kinetics model to calculate delayed neutron power as a function of time. Both limit the shutdown margin to the technical specification shutdown margin limit, 1 percent *K/K. Revision 0 calculated the shutdown reactivity but limited the amount to the technical specification minimum shutdown margin. In Revision 1 the thermal feedback calculation has been deleted and the shutdown margin of 1 percent *K/K is used.

Since the amount of shutdown reactivity has not changed, the NRC staff determined that the proposed revision is acceptable.

37. Section 4.3.1: Since no thermal feedback is modeled (See Item 36), the total rod worth is set to the technical specification shutdown margin value.

Since the amount of shutdown reactivity has not changed, the NRC staff finds this change acceptable.

38. Section 4.3.1: A new section, "Fission Products and Decay Heat," is added to Revision 1 to describe the decay heat modeling. ANSI/ANS Standard 5.1-1979 is used with a $2\sigma$ uncertainty.

ANSI/ANS Standard 5.1-1979 has been used acceptably for the determination of decay heat and the NRC staff determined it is acceptable to use in this case.

39. Section 4.3.2, "Reactor Building Model," is deleted in its entirety and replaced with a new subsection titled "Containment Boundary Conditions," since the model is no longer used in the Revision 1 methods. This section describes the use of FATHOMS to determine two boundary conditions for RELAP5 mass and energy release calculations. These two boundary conditions are: (1) the containment backpressure against the LOCA break flow discharges and (2) the sump water temperature used for the ECCS suction supply after transition to the sump recirculation mode. These quantities were determined by a RELAP5 containment model in Revision 0.

Revision 1 states that the boundary conditions are based on FATHOMS cases that are consistent with the RELAP5 analysis in terms of break size and intended application. Since FATHOMS calculates containment pressure and sump temperature conservatively, this change is acceptable.

40. Section 4.3.2: FATHOMS calculates boundary conditions for RELAP5 small break LOCA mass and energy release calculations. In addition to those boundary conditions listed in Item 39, spilled HPI flow is also modeled in FATHOMS for input into RELAP5 for small break LOCA mass and energy release calculations.

See Item 34.

41. Section 4.3.2: Revision 1 revises the modeling of emergency feedwater (EFW) actuation and control. The revision makes the modeling consistent with current station procedures and the UFSAR Chapter 15 small break LOCA peak cladding temperature analysis. The delay from reactor trip to EFW actuation is changed from 10 minutes to 20 minutes. The steam generator level is controlled to the loss of subcooled margin (LSCM) setpoint. Revision 0 specified the value of this setpoint. Revision 1 does not specify the value, only that the LSCM setpoint is used and must be adjusted for level instrument uncertainty.

The NRC staff determined that this change is consistent with current station procedures and the small break LOCA peak cladding temperature analysis and

conservatively accounts for instrument uncertainty. Therefore, it is acceptable to the NRC staff.

42. Section 4.3.2: The modeling of the steam generator pressure control is revised to change the time at which the atmospheric dump valves are credited and to change the target cooldown rate. An operator is assumed to begin manipulating these valves locally beginning at 60 minutes, rather than the previous time of 30 minutes, after break initiation. The cooldown rate is revised from 80 °F/hr to > 50 °F/hr, based upon the core exit temperature.

The NRC staff determined that these changes are consistent with current station procedures and are therefore acceptable.

43. Section 4.3.2: The modeling of the emergency feedwater system is revised to credit only one motor-driven emergency feedwater pump initiating at 20 minutes and feeding only one steam generator.

The single failure of a 4160 V switchgear results in the loss of one motor-driven emergency feedwater pump. No credit is taken for the turbine-driven emergency feedwater pump. Revision 1 states that this change is consistent with current station procedures and the small break LOCA peak cladding temperature analysis. The change is, therefore, acceptable to the NRC staff since it more accurately reflects licensing bases limitations. The NRC staff notes that taking no credit for the turbine driven emergency feedwater pump is consistent with the licensing basis for the MSLB detection and feedwater isolation system.

44., 45., 46. Editorial

Chapter 5 Steam line Break Mass and Energy Release Analyses

47., 48, 49., 50., 51., 52. Editorial

Chapter 6 Containment Analysis

Revision 1 proposes the use of two containment thermal-hydraulic analysis computer codes. The use of FATHOMS was proposed in Revision 0 and approved by the NRC in a letter dated March 15, 1995. Revision 1 proposes some changes to this analysis that are discussed below.

Revision 1 also proposes the use of the GOTHIC 7.0 computer code for Oconee containment thermal-hydraulic calculations. GOTHIC 7.0 was developed by Numerical Applications, Inc., for EPRI. GOTHIC 7.0 contains state-of-the-art models for the phenomena expected to occur during a high energy line break (LOCA or MSLB) in a nuclear reactor containment.

GOTHIC 7.0 has been validated against an extensive data base including analytical solutions and data from separate effects and integral tests[9].

[9]GOTHIC Containment Analysis Package: Qualification Report Version 7.0 July 2001

The licensee's July 7, 2003, letter provides a summary of the major differences between GOTHIC 7.0 and FATHOMS that are relevant to the licensee's application of GOTHIC 7.0 to Oconee. These include the modeling of drops and the modeling of interphase heat transfer.

The licensee's July 7, 2003, letter describes the modeling of the blowdown flow in GOTHIC 7.0. Separate flowrates and enthalpies for the steam and the liquid phases of the blowdown are calculated in the RELAP5 code and passed to GOTHIC 7.0 as separate boundary conditions. In GOTHIC 7.0 all liquid flow enters as droplets when the RCS pressure exceeds the containment pressure.

While FATHOMS solves one energy equation for the vapor phase (air and steam) and another for the liquid phase (drops and liquid), GOTHIC 7.0 models the drops and liquid within a computational control volume so that each can have a different temperature.

The GOTHIC 7.0 drop liquid conversion option signals the use of the GOTHIC 7.0 default models for drop behavior including entrainment, agglomeration, and deposition. The calculated bulk average velocities of the Oconee containment atmosphere are not great enough to cause entrainment. GOTHIC 7.0 does model the dripping of condensate from ceilings and equipment (1/6th of the condensation rate for the volume). The July 7, 2003, letter states that this dripping model was included to improve agreement with experimental data.

The licensee's July 7, 2003, letter states that the Uchida condensation heat transfer coefficient will be used for all heat slabs in GOTHIC 7.0 calculations. The Uchida correlation is recommended by ANSI/ANS Standard 56.4-1983, "Pressure and Temperature Transient Analysis for Light Water Reactor Containments." While Oconee is not considered to be an SRP plant, the use of the Uchida correlation is consistent with the guidance in SRP 6.2.1.1.A, "PWR Dry Containment, Including Subatmospheric Containments." The GOTHIC Uchida model was also previously accepted by the staff in approving Revision 0 and as part of the review of the application of GOTHIC 6.0 to Kewaunee (ADAMS Accession No. ML012490176). It is considered conservative by the NRC staff and is typically used for licensing calculations.

Revision 0 notes that ANSI/ANS Standard 56.4-1983 allows the Tagami and Uchida condensation heat transfer correlations. The Tagami correlation applies to blowdown conditions. However, since the Tagami correlation reverts to the Uchida correlation after blowdown, Revision 0 uses the Uchida correlation throughout. This remains the case in Revision 1. The NRC staff determined that this is conservative since the Tagami correlation typically predicts a higher rate of heat transfer.

The Oconee reactor building is modeled as two volumes; the containment air volume and a separate sump volume that consists of the lowest 6.35 feet of the reactor building. The two nodes are referred to as the atmosphere region and the sump region, respectively. The noding is analogous to the CONTEMPT[10] containment model except that they are separate volumes in FATHOMS whereas the atmosphere and sump regions are part of the same volume in CONTEMPT.

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[10]Hargroves, Don W., et al, CONTEMPT-LT/028 - A Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident, USNRC NUREG/CR-0255 March 1979.

Following a LOCA, the nitrogen cover gas in the core flood tanks is released to the containment through the break. The addition of this gas is, in general, conservative. However, the July 7, 2003, letter points out that because the peak containment pressure is reached prior to the time at which the nitrogen gas is injected into the containment for all cases analyzed, it has no impact on the calculated peak containment pressure for Oconee.

GOTHIC 7.0 offers other options for calculating condensation heat transfer. These have not been reviewed by the NRC staff for application to Oconee (e.g., the mist diffusion layer model and the Gido-Koestel correlation) and shall not be used for design basis licensing calculations.

The July 7, 2003, letter provided comparisons of GOTHIC 7.0 containment LOCA and MSLB calculations with those of FATHOMS. These comparisons show that:

GOTHIC 7.0 predicts LOCA peak pressures that are generally slightly lower (approximately 0.5 psi)

GOTHIC 7.0 predicts LOCA temperatures that are generally slightly lower (approximately 0.5 °F)

GOTHIC 7.0 predicts MSLB pressures that are generally slightly higher (approximately 0.5 psi)

GOTHIC 7.0 predicts MSLB peak temperatures that are generally lower (approximately 10 °F)

GOTHIC 7.0 predicts sump water volumes that are generally lower for both the LOCA and MSLB

GOTHIC 7.0 predicts sump water temperatures that are generally lower for both the LOCA and MSLB

The July 7, 2003, letter explains these trends as follows. For the LOCA events, both codes predict a saturated containment atmosphere due to the large amount of drops in the containment atmosphere. The lower containment pressure and temperature predicted by GOTHIC 7.0 are due to additional drop holdup in the containment atmosphere. The effective increase in the heat capacity of the atmosphere due to the difference in drops in the atmosphere explains the reduction in containment temperature compared with FATHOMS.

For the MSLB events, the higher pressure predicted by GOTHIC 7.0 is due to additional steam generation from the improved interface heat and mass transfer models and from the drops that are generated due to dripping. The additional vaporization takes heat from the atmosphere and reduces the atmosphere temperature.

With the restriction on the use of condensation heat transfer options, the NRC staff finds the use of GOTHIC 7.0 acceptable as described in Revision 1 and as discussed below.

53. Section 6.1: For the long term LOCA analyses, the large break and small break containment calculations are divided into two time segments. They were previously divided into three.

The large break and small break mass and energy release calculations are divided into two segments rather than three. The two time segments are: (1) ECCS injection flow coming from the BWST, and (2) after switchover to sump recirculation. In order to be consistent with this division, the containment analysis is also divided into two time segments so that they can be modeled with the same boundary conditions. The NRC staff determined that this approach makes the analysis more consistent and is, therefore, acceptable.

54. Section 6.2.3: The blowdown mass flow is assumed to consist of droplets. The size of these droplets is assumed to be 100µ[11].

The GOTHIC computer code assumes the liquid portion of the break flow consists of droplets and recommends a drop size of 100µ based on spray tests reported in the Journal of the American Institute of Chemical Engineering, Volume 8, No. 2, 1962. This is acceptable to the NRC staff based on the following:

During a DBE LOCA, the water entering the containment as RCS break flow is at a temperature above the saturation temperature at the containment pressure. Upon entering the containment, a portion of the water flashes to steam, fracturing the water jet into fine droplets. The experimental data cited above have shown that when superheated water flashes to steam, the mean drop diameter is less than 100µ.

The GOTHIC 7.0 qualification analyses, presented in the GOTHIC 7.0 code documentation qualification report, were performed using a drop diameter of 100µ. These qualification analyses showed that GOTHIC 7.0 calculations, with the 100µ assumption, agreed with, and typically bounded, the measured pressure and temperature response from blowdown tests and measured pressure drops from orifice pressure drop tests.

A 100µ drop has a terminal velocity (rainout velocity) of between 1 and 2 ft/sec. This is a realistic terminal velocity and allows for the break drops to be in the containment atmosphere for a realistic time.

Therefore, the use of the 100µ drop size is acceptable.

55. Section 6.3.1: The RELAP5 analysis now runs through to the end of the injection phase rather than stopping at 1800 seconds.

See Item 12.

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[11] µ represent "micron," a millionth of a meter

56. Section 6.3.2: The RELAP5 analysis now runs through to the end of the injection phase rather than stopping at 1800 seconds. Also there are two time segments instead of three.

> See Item 12.

57., 58., 59. Section 6.3.3: The large break and small break containment calculations are divided into two time segments. They were previously divided into three.

> See Item 53.

60. Section 6.3.3: The blowdown mass flow is assumed to consist of droplets. The size of these droplets is assumed to be 100µ.

> See item 54.

61. Section 6.3.3: The LPI flowpaths and mixing volumes are revised.

> The LPI flowpath is revised to reflect the fact that for some failure scenarios, it is possible for only one LPI cooler to be supplied with cooling water although both will have LPI flow. The current model combines both LPI trains into one equivalent train. The NRC staff found that the proposed revision more accurately models the LPI system and is, therefore, acceptable.

62. Through 69. Revision 1 describes changes made to the noding of the FATHOMS model.

> The NRC staff did not review the details of the revised noding. The NRC staff determined that the adequacy of the noding can be judged by the reasonableness of the results. On the basis of reasonable results, the noding is acceptable.

70. Section 6.4.3: Revision 1 revises the modeling of RCS and secondary side heat structures.

> The change is proprietary. The NRC staff determined that the change makes the accounting of heat structures more complete and is, therefore, acceptable.

71. Section 6.4.3: The droplet size of the blowdown mass is changed to 100µ.

> See item 54.

72. Section 6.4.5: The noding of the small break FATHOMS model has been revised.

> The NRC staff did not review the detailed noding of the FATHOMS model. The NRC staff determined that the adequacy of the noding can be judged by the reasonableness of the results. On the basis of reasonable results, the noding is acceptable.

73. Section 6.4.5: Revision 1 discusses a boundary condition for a break in a high pressure injection line. A model for the BWST is eliminated from FATHOMS. The NRC staff determined that the event is adequately modeled without modeling the BWST. Therefore, this change is acceptable.

74. Section 6.6 Editorial

75. Chapter 7 Editorial

76. Supplement 1 Editorial

Additional revisions to Revision 1 identified in the July 7, 2003, letter from Duke Energy Corporation:

The long-term mass and energy release boundary condition modeling in FATHOMS has been installed in GOTHIC 7.0. This modeling approach enables the interpolation of the BFLOW code results (for mass and energy release after blowdown) as an input boundary condition. It is described in Sections 2.2.1.1, 2.4.1.2, and 6.3 of Revision 1.

This modification is made in lieu of using the control variables available in GOTHIC 7.0. The proposed modeling is similar to that used in FATHOMS that was approved by the NRC's March 15, 1995, SER and is acceptable.

Because GOTHIC 7.0 is being modified to accept BFLOW input, the licensee's version of GOTHIC 7.0 will be uniquely identified as GOTHIC 7.0/DUKE.

Environmental Qualification

The temperature envelope curve for environmental qualification was provided in a July 28, 2003, letter. The GOTHIC 7.0 results are compared to the FATHOMS curves for containment atmosphere temperature, pressure, and the mass weighted liquid temperature, which is a combination of the liquid and droplet phases. The NRC staff found that the agreement is good. The differences are explainable in terms of the difference in modeling between the two codes.

## 4.0 CONCLUSION

Duke topical report DPC-NE-3003-P, "Mass and Energy Release and Containment Methodology," has been revised. Two new computer codes are proposed for Revision 1: RETRAN-3D will replace RETRAN-02 for steam line break mass and energy release calculations and the GOTHIC 7.0 computer code will be used in addition to the FATHOMS code approved as part of DPC-NE-3003-P for containment thermal hydraulic calculations. Both new codes upgrade the calculation capabilities of the codes they replace. (RETRAN-3D has been approved by the NRC in Enclosure 1 that precedes this SE.)

DPC-NE-3003-P, Revision 1, proposes to use GOTHIC 7.0 with conservative input and heat transfer correlations. For these reasons, the NRC staff finds the use of GOTHIC 7.0 to be acceptable. GOTHIC 7.0 offers options for calculating condensation heat transfer other than

the Uchida correlation proposed in DPC-NE-3003-P, Revision 1. These correlations have not been reviewed by the NRC for application to Oconee (e.g., the mist diffusion layer model and the Gido-Koestel correlation) and shall not be used for design basis licensing calculations.

In addition, many changes are made in Revision 1 to calculation assumptions to reflect changes in plant operation and procedures. These changes ensure that the safety calculations will accurately predict the plant behavior following a design basis accident and that the relevant safety criteria will not be exceeded.

The changes made in Revision 1 comply with the AEC General Design Criteria 10, 15, and 49, published in the Federal Register of July 11, 1967, which apply to Oconee.

The methods described in Revision 1 may be used to perform analyses in support of licensing applications related to containment accident response for peak containment pressure and temperature and environmental qualification.

Principal Contributor: R. Lobel

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