August 30, 2006

Mr. David A. Christian
Senior Vice President
and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: KEWAUNEE POWER STATION (KEWAUNEE), MILLSTONE POWER STATION, UNIT NOS. 2 AND 3 (MILLSTONE 2 AND 3), NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2 (NORTH ANNA 1 AND 2), AND SURRY POWER STATION, UNIT NOS. 1 AND 2 (SURRY 1 AND 2) - APPROVAL OF DOMINION'S TOPICAL REPORT DOM-NAF-3, "GOTHIC METHODOLOGY FOR ANALYZING THE RESPONSE TO POSTULATED PIPE RUPTURES INSIDE CONTAINMENT" (TAC NOS. MC8831, MC8832, MC8833, MC8834, MC8835, AND MC8836)

Dear Mr. Christian:

By letter dated November 1, 2005, as supplemented by letters dated June 8 and July 14, 2006, Dominion Energy Kewaunee, Inc., Dominion Nuclear Connecticut, Inc., and Virginia Electric and Power Company, (the licensees), requested approval for the generic application of Topical Report DOM-NAF-3, "GOTHIC Methodology for Analyzing the Response to Postulated Pipe Ruptures Inside Containment."

GOTHIC (Generation of Thermal-Hydraulic Information for Containments) is a general purpose thermal-hydraulics computer code developed by the Electric Power Research Institute for performing containment analyses. The licensees have developed an analytical method using the GOTHIC methodology to replace the current containment analysis at Kewaunee, Millstone 2 and 3, North Anna 1 and 2, and Surry 1 and 2.

The enclosed Safety Evaluation (SE) documents the basis for the U.S. Nuclear Regulatory Commission (NRC) staff's conclusion's that Topical Report DOM-NAF-3 is acceptable for the licensees' nuclear facilities. The SE defines the basis for the acceptance of the report.

In accordance with the guidance provided on the NRC website, the licensees are requested to publish an accepted version of this topical report within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE between the title page and the abstract. It must be well indexed such that information is readily located. Also, it must contain, in appendices, historical review information, such as questions and accepted responses, and original report pages that were replaced. The accepted version shall include an "-A" (designated accepted) following the report identification symbol.

If the NRC staff's criteria or regulations change such that its conclusions as to the acceptability of the topical report are invalidated, then these licensees will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

Sincerely,

/RA/

Ho K. Nieh, Acting Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket Nos. 50-305, 50-336, 50-423,
50-338, 50-339, 50-280, and 50-281

Enclosure:
Safety Evaluation

cc w/encl: See next page

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ADAMS Accession No. ML062420511 *date of memo transmitting safety evaluation

OFFICE	NRR/LPL2-1/PM	NRR/LPD2-1/LA	NRR/LPL1-2/PM	NRR/SCVB/BC	NRR/LPL2-1/BC	NRR/DPR/D
NAME	SMonarque	MO'Brien	VNerses GMiller for	RDennig	EMarinos	HNeih
DATE	08/16/2006	08/24/2006	08/21/2006	08/01/2006*	08/22/2006	08/30/2006

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08/24/2006

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO TOPICAL REPORT DOM-NAF-3

KEWAUNEE POWER STATION (KEWAUNEE)

MILLSTONE POWER STATION, UNIT NOS. 2 AND 3 (MILLSTONE 2 AND 3)

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2 (NORTH ANNA 1 AND 2)

SURRY POWER STATION, UNIT NOS. 1 AND 2 (SURRY 1 AND 2)

DOCKET NOS. 50-305, 50-336, 50-423, 50-338, 50-339, 50-280, AND 50-281

1.0 INTRODUCTION

By letter dated November 1, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML053060266 (pages 1 through 40) and ML053060273 (pages 41 through 85)), as supplemented by letters dated June 8 and July 14, 2006 (ADAMS Accession Nos. ML062070314 and ML062020394, respectively), Dominion Energy Kewaunee, Inc., Dominion Nuclear Connecticut, Inc., and Virginia Electric and Power Company (the licensees), requested approval for the generic application of Topical Report DOM-NAF-3, "GOTHIC Methodology for Analyzing the Response to Postulated Pipe Ruptures Inside Containment." The licensees requested the U.S. Nuclear Regulatory Commission (NRC) staff's approval of this topical report to perform licensing basis analyses for the containment response for pressurized-water reactors (PWRs) with large, dry containments. The June 8, 2006, letter responded to the NRC staff's request for additional information, dated April 28, 2006 (ADAMS Accession No. ML061180146). The July 14, 2006, letter corrected a modeling error identified by the licensees, and provided additional information requested by the NRC staff.

GOTHIC (Generation of Thermal-Hydraulic Information for Containments) is a general-purpose thermal-hydraulics code for containment analysis developed for the Electric Power Research Institute (EPRI) by Numerical Applications, Inc. (NAI), for applications in the nuclear power industry. This safety evaluation (SE) addresses the licensees' proposed use of GOTHIC for licensing basis analyses. Specifically, GOTHIC methodology would be used to replace the evaluation methods in the updated final safety analysis reports (UFSARs) for the containment design requirements listed below:

1. Loss-of-coolant accident (LOCA) containment peak pressure and temperature

2. Main steam line break (MSLB) containment peak pressure and temperature

3. LOCA containment depressurization time (CDT) for Surry 1 and 2 and North Anna 1 and 2
4. LOCA containment subatmospheric peak pressure (SPP) for Surry 1 and 2 and North Anna 1 and 2

5. Net positive suction head available (NPSHA) for pumps that take suction from the containment sump. For Surry 1 and 2 and North Anna 1 and 2, a time-dependent NPSHA is calculated from a transient containment response for the inside recirculation spray (IRS), outside recirculation spray (ORS), and low head safety injection (LHSI) pumps

6. Minimum and maximum sump water level and liquid temperature for input to other analyses (e.g. , strainer debris head loss and component stress analyses)

7. Containment liner temperature verification

8. Equipment qualification (EQ) temperature validation, and

9. Transient performance of closed cooling loops for heat exchangers associated with the emergency core cooling systems (ECCS) and containment heat removal systems.

As stated in the licensees' application and discussed in Section 3.0 below, GOTHIC methodology for some of the above proposed design-basis analyses has been previously approved by the NRC staff for other licensees. Therefore, the primary focus of this SE will be on the proposed use of GOTHIC for applications that have not been previously approved by the NRC staff; and, hence, could not be implemented by the licensees using the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.59.

2.0 REGULATORY EVALUATION

The General Design Criteria (GDC) contained in 10 CFR Part 50, Appendix A (as stated below), establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants. The NRC staff considered the following requirements for this review.

> Criterion 4, *Environmental and dynamic effects design bases*. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

> Criterion 16, *Containment design*. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

> Criterion 38, *Containment heat removal*. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.
>
> Criterion 50, *Containment design basis*. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

The NRC staff used the guidance in the Standard Review Plan (SRP), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," NUREG-0800, Section 6.2.1, "Containment Functional Design," Section 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments," Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," and Section 6.2.2, "Containment Heat Removal Systems," for this review.

The NRC staff also used Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 3, November 2003, and NUREG-588, "Interim Staff Position on Equipment Qualification of Safety-Related Electrical Equipment," Revision 1, November 1980 as additional guidance for its review.

3.0 TECHNICAL EVALUATION

GOTHIC solves the conservation equations for mass, momentum and energy for multi-component, multi-phase flow in lumped parameter and/or multi-dimensional geometries. The phase balance equations are coupled by mechanistic models for interface mass, energy and momentum transfer that cover the entire flow regime from bubbly flow to film/drop flow, as well as single phase flows. The interface models allow for the possibility of thermal non-equilibrium between phases and unequal phase velocities, including countercurrent flow. GOTHIC includes full treatment of the momentum transport terms in multidimensional models, with optional models for turbulent shear and turbulent mass and energy diffusion. Other phenomena include models for commonly available safety equipment, heat transfer to structures, hydrogen burn and isotope transport.

GOTHIC is maintained by EPRI under a 10 CFR Part 50, Appendix B quality assurance program, is widely used in the U.S. and worldwide, and has been extensively verified and validated by NAI, as documented in the GOTHIC Qualification Manual.[1] The licensees have indicated that they have participated in the EPRI GOTHIC Advisory Group since the late 1980s in order to ensure a solid understanding of the code capabilities and limitations, to monitor industry applications, and to guide the code qualification effort.

For Topical Report DOM-NAF-3, the licensees used GOTHIC Version 7.2dom, which consists of the EPRI-released Version 7.2 and two enhancements specific to the licensees that were implemented during testing of the GOTHIC containment model for Surry 1 and 2. As noted above, the NRC staff has performed similar reviews for GOTHIC methodology. Recently this included the use of GOTHIC Version 7.0 for Ft. Calhoun[2] and Kewaunee[3], and GOTHIC Version 7.1 for Framatome Advanced Nuclear Power (ANP), Inc.[4] The differences between GOTHIC 7.0, 7.1, 7.2, and 7.2dom, with respect to the analyses of the containment response to design-basis accidents (DBAs) as discussed in this SE are not significant. For the most part, the later versions correct coding errors and include user features to enable the user to apply models consistent with the NRC staff's limitations. For example, in GOTHIC Version 7.2, the Mist Diffusion Layer Model (MDLM) heat and mass transfer option was replaced with the Diffusion Layer Model (DLM) option and optional enhancement factors for mist generation and film roughening effects. The DLM option eliminated the boundary layer mist formation and the height dependent film roughness enhancements to address concerns identified during the NRC staff's review of the Kewaunee amendment (see footnote 3).

In Section 3.0 of DOM-NAF-3, the licensees provided the proposed methodology for constructing GOTHIC models to perform licensing basis analyses for large, dry containments. The licensees stated that the methods are intended to provide realistic but conservative results based on previously accepted PWR containment methodologies and the extensive validation base for GOTHIC. In Section 4, the licensees documented GOTHIC containment analyses for Surry 1 and 2 that demonstrated the acceptability of the analysis methodology described in Section 3. Analyses were performed for LOCA peak pressure and temperature, MSLB peak pressure and temperature, containment depressurization, and NPSHA for the LHSI pumps. Benchmark comparisons were made to the LOCTIC analyses described in the Surry 1 and 2 UFSAR. As described in UFSAR Chapter 14.B.2.3.3.1 for Surry 1 and 2, LOCTIC is a computer program used to calculate containment pressure and temperature transients.

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[1] NAI 8907-09 Rev 8, "GOTHIC Containment Analysis Package Qualification Report, Version 7.2," published by EPRI, September 2004,

[2] ADAMS Accession No. ML033100290, letter from A. B. Wang, USNRC, to R. T. Ridenoure, Omaha Public Power District, "Fort Calhoun Station, Unit No. 1 - Issuance of Amendment (TAC No. MB7496)," dated November 5, 2003.

[3] ADAMS Accession No. ML032681050, letter from A. C. McMurtray, USNRC, to T. Coutu, Nuclear Management Company, LLC, "Kewaunee Nuclear Power Plant - Issuance of Amendment (TAC No. MB6408)," dated September 29, 2003.

[4] ADAMS Accession No. ML052240302, Letter from H. N. Berkow, USNRC, to R. L. Gardner, Framatome, "Final Safety Evaluation for Framatome ANP Topical Report BAW-10252(P), Revision 0, 'Analysis of Containment Response to Postulated Pipe Ruptures Using GOTHIC,' (TAC No. MC3783)," August 31, 2005.

Although not documented in Topical Report DOM-NAF-3, the licensees indicated that the bench-marking also included GOTHIC model adjustments to mimic the same physical behavior as LOCTIC. For example, the GOTHIC droplet phase was effectively disabled to support a comparison to the LOCTIC equilibrium flash model and the containment volume liquid/vapor interface area was set to zero. The licensees stated that these benchmarks used long-term mass and energy data calculated by LOCTIC. The licensees' objective was to demonstrate adequate modeling of containment components, nodalization of piping systems, and modeling of spray systems, with respect to another containment response code. The licensees confirmed that these benchmarks showed a successful comparison of the containment response.

The licensees have also performed a sensitivity study for break locations, single failures, and design inputs to determine conservative assumptions for each required analysis for Surry 1 and 2. The results are contained in Table 4.7-1 of Topical Report DOM-NAF-3 and are consistent with the current LOCTIC analyses for Surry 1 and 2 with the exception of the limiting single failure for the calculation of NPSHA for the ORS and IRS pumps. Since each plant has specific design criteria and engineered safety features that require sensitivity studies, the licensees have stated that they will perform similar bench marking and sensitivity studies to define the set of conservative assumptions for the other plants, as part of the licensing basis methodology conversion.

The licensees' demonstration analysis and bench marking for Surry 1 and 2 provided reasonable justification for the appropriateness of its proposed GOTHIC methodology. In the following sections, specific components of Topical Report DOM-NAF-3 methodology are discussed further beginning with features that have been previously approved by the NRC staff for similar applications.

3.1 Containment Response Methodology for DBAs

As noted above, the NRC staff has previously approved GOTHIC methodologies for analyzing containment response to LOCA and MSLB events (see footnotes 2, 3, and 4). The analyses use models to maximize containment pressure and temperature using inputs to the GOTHIC methodology mass and energy release data that are generated by other NRC staff-approved methods. In response to the NRC staff's request for additional information, the licensees have confirmed that the DOM-NAF-3 methodology for maximizing LOCA and MSLB containment pressure and temperature uses NRC staff-approved models for the containment response (e.g., the Direct/DLM for heat transfer between passive heat sinks and the containment atmosphere in Topical Report DOM-NAF-3, Section 3.3.2, and the break release droplet model with 100-micron droplets in Topical Report DOM-NAF-3, Section 3.5.1). This aspect of Topical Report DOM-NAF-3 (Applications 1-4, Section 1.0) is acceptable to the NRC staff and no further review is required.

3.2 Post-Reflood Mass and Energy Release Model

The NRC staff has also previously reviewed and approved GOTHIC methodology for post-reflood mass and energy release calculation for Framatome ANP (see footnote 4). However, in response to the NRC staff's request for additional information, the licensees stated that they were unable to make full comparison with Framatome's methodology because it contained proprietary information. The NRC staff has identified certain differences between Framatome's and the licensees' methodologies with regard to their approach for major component modeling, but the basic elements of both methodologies for long-term (post-reflood) mass and energy release calculation are similar. For both methodologies the transition time for GOTHIC generated mass and energy calculation starts at the end of reflood, once the core is quenched and has been fully covered with water, and ECCS injection maintains the core covered so that decay heat removal and sensible heat removal is assured at all times. Both methodologies account for all remaining stored energy in the primary and secondary systems in accordance with SRP 6.2.1.3 for the post-reflood phase.

The licensees' GOTHIC methodology for long-term mass and energy release acquires the energy for each source term at the end of reflood from the fuel vendor's mass and energy release analysis. The rate of mass and energy release is determined by a simplified GOTHIC reactor coolant system (RCS) model that is coupled to the containment volume. Thus, the flow from the vessel to the containment is dependent on the GOTHIC-calculated containment pressure. Lumped volumes are used for the vessel, down-comer, cold legs, steam generator secondary side, up-flow portion of the steam generator tubes and down-flow portion of the steam generator tubes. Separate sets of loop and secondary system volumes are used for the intact and broken loops with the connections between the broken loop and containment as necessary for the modeled break location.

In Section 4.3.2 and 4.4.2 of Topical Report DOM-NAF-3, the licensees provided comparison of mass and energy release data calculated by the proposed simplified GOTHIC RCS model with data from the NRC staff-approved FROTH methodology in WCAP-8264-P-A[5] and WCAP-10325-P-A[6], as implemented using the Stone & Webster (SWEC) LOCTIC containment response code. For the hot leg break case, the GOTHIC integral mass release matches closely with the FROTH/LOCTIC generated mass release, while the GOTHIC integral energy release was slightly higher and more conservative than the FROTH/LOCTIC generated energy. For the pump suction break case, both the integral mass and energy releases match very closely with the FROTH/LOCTIC generated data.

Although this comparison shows that no margin was gained with the proposed methodology, with respect to mass and energy releases, the simplified RCS methodology provides a reduction in containment depressurization time and a less severe pressure increase following containment spray termination, as shown in Section 4.4 of Topical Report DOM-NAF-3. The licensees attribute this gain in margin to other mechanistic features of GOTHIC that were previously reviewed and approved by the NRC staff. The NRC staff concurs with this assessment and finds the methodology for post-reflood mass and energy release calculation acceptable. However, the modeling technique is highly complex and iterative (e.g. modeling of the primary metal stored energy); therefore, as a condition of approval for Topical Report DOM-NAF-3, conservative mass and energy release values calculated for Surry 1 and 2 shall be duplicated for North Anna 1 and 2, Millstone 2 and 3, and Kewaunee through appropriate bench marking and model adjustment prior to implementing this methodology in licensing applications.

3.3 Methodology for Calculating NPSHA

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[5] WCAP-8264-P-A, Rev. 1, 'Westinghouse Mass and Energy Release Data for Containment Design," August 1975. (WCAP-8312-A is the Non-Proprietary version).

[6] WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," May 1983. (WCAP-10326-A is the Non-Proprietary version.)

Section 3.8 of Topical Report DOM-NAF-3 describes the licensees' proposal to perform transient calculation of NPSHA through conservative model adjustment of the long-term containment response model. The calculation is performed internally in GOTHIC using an industry standard formulation for prediction of pump NPSHA. The same formula was used previously in the SWEC LOCTIC containment analysis methodology, which performed a transient calculation of NPSHA for the current licensing bases[7] at North Anna 1 and 2 and Surry 1 and 2. NPSHA is the difference between the fluid stagnation pressure and the saturation pressure at the pump intake. NPSHA depends directly on transient predictions of sump temperature, sump water level, and containment pressure.

The licensees intend to employ this methodology for North Anna 1 and 2 and Surry 1 and 2. Both plants have subatmospheric containments that are required to be depressurized following a DBA in accordance with the assumptions in the dose consequence analyses. The current licensing bases for North Anna 1 and 2 and Surry 1 and 2 allow credit for containment over pressure to calculate NPSHA for the the IRS, ORS, and LHSI pumps[8] [9]. Although the proposed methodology is applicable to any large, dry containment, it cannot be used for the other licensees' plants that do not credit containment overpressure to calculate NPSHA in their licensing bases.

In the licensees' proposed methodology, the GOTHIC simplified RCS containment model is used with a separate small volume for the pump suction. The pump suction volume elevation and height are set so that the mid-elevation of the volume is at the elevation of the pump first-stage impeller centerline. The volume pressure, with some adjustments for sump depth, is used in the NPSHA calculation. The temperature in the suction volume provides the saturation pressure. The junction representing piping between the sump and the suction volume reflects the friction pressure drop between the sump and the pump suction. A correlation is used to define the sump depth or liquid level as a function of the water volume in the containment. The correlation accounts for the sump geometry variation with water depth and accounts for the holdup of water in other parts of the containment.

The proposed methodology incorporated several adjustments to the simplified RCS containment model to ensure a conservative calculation of NPSHA. A multiplier of 1.2 is applied to the heat transfer coefficient for the containment heat sinks to compensate for the non-conservative values (with respect to NPSHA calculation) generated by the Direct DLM heat

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[7] ADAMS Accession No. 9811090068, Letter from J. P. O'Hanlon (VEPCO) to USNRC, "Virginia Electric and Power Company, North Anna and Suny Power Stations Units 1 and 2, Generic Letter 97-04 - Assurance of Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps; Response to a Request for Additional Information," Serial No. 98-546, October 29, 1998.

[8] ADAMS Accession No. 9903030158, Letter from N. Kalyanam (USNRC) to J. P. O'Hanlon (VEPCO), "Completion of Licensing Action for Generic Letter 97-04, 'Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps'; North Anna Power Station, Unit Nos. 1 and 2 (TAC Nos. MA0015 and MA0016)," February 25, 1999.

[9] ADAMS Accession No. 9904070170, Letter from G. E. Edison (USNRC) to J. P. O'Hanlon (VEPCO), "Completion of Licensing Action for Generic Letter 97-04, 'Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps'; Suny Power Station, Unit Nos. 1 and 2 (TAC Nos. MA0050 and MA0051)," April 1, 1999.

transfer Model. All of the spray water is injected as droplets into the containment atmosphere (nozzle spray flow fraction of 1). Analyses are performed using the largest Sauter spray droplet size and a confirmatory analysis is performed by reducing the Sauter diameter by 2, which sufficiently covers code and spray performance uncertainty without creating drops too small that may cause excess droplet holdup in the atmosphere. A conservative water holdup volume is subtracted from the containment liquid volume to reduce the sump water height. Other adjustments include use of upper limit for containment free volume and minimum initial containment pressure. The conservatism incorporated in this methodology meets the applicable regulatory positions in RG 1.82

In Section 4.5 of Topical Report DOM-NAF-3, the licensees provided benchmark results comparing GOTHIC calculation of LHSI pump NPSHA to LOCTIC analyses from the UFSAR for Surry 1 and 2 for a pump suction break LOCA transient. The GOTHIC results showed good agreement with the LOCTIC case. The more realistic GOTHIC modeling of the RCS and steam generators resulted in slightly more energy being transferred to the containment at the time the LHSI pumps take suction from the sump. At the time of minimum NPSHA, the GOTHIC sump temperature is actually slightly higher than the LOCTIC value; however, the GOTHIC pressure is also higher, yielding a small, net increase in NPSHA. The licensees concluded that the higher sump temperature and containment pressure than LOCTIC is consistent with the additional energy addition from the RCS model, and is considered to be a reasonable and more accurate system response.

The proposed use of GOTHIC methodology to calculate NPSHA uses an industry standard formulation that was previously approved by the NRC staff and incorporates applicable conservatisms contained in RG 1.82. As such, the NRC staff finds this acceptable.

3.4 GOTHIC Application for Component Design Verification

The NRC staff's previous acceptance of the GOTHIC containment response calculation methodologies for containment design limits does not explicitly cover the use of GOTHIC results for component design verification. As a result, in Section 2.3 of Topical Report DOM-NAF-3, the licensees included Applications 6-9 for the NRC staff to review and approve regarding the use of GOTHIC output for specific component analyses.

3.4.1 Application 6: Sump Data for Input to Other Analyses

GOTHIC modeling assumptions can be biased to produce conservative results with respect to sump water level and liquid temperature. The licensees' requested approval to use these conservative results for validation against component design limits. As discussed in Section 3.3, the methodology for performing pump NPSHA calculations produces a higher sump water temperature profile than LOCTIC and is thus more conservative than LOCTIC. The licensees' plan to use this GOTHIC sump water temperature profile for validation against component design limits.

Because the licensees are using a sump water temperature profile that is more conservative than the NRC staff-approved LOCTIC code, the NRC staff finds the use of the GOTHIC generated sump temperature and level data for input to other analyses acceptable.

3.4.2 Application 7: Containment Liner Temperature Verification

The licensees' proposed methodology for the containment liner temperature verification is a sightly modified version of the peak containment temperature model. A conservative containment liner response is obtained by adding a small conductor that has the same construction and properties as the liner conductor. A conductor surface area of 1 ft^2 is used to minimize impact on the lumped containment pressure and temperature response. The inside heat transfer option is the same as used for the actual liner conductor (Direct with DLM) with a multiplier of 1.2 for conservatism.

The Direct/DLM model has been previously accepted by the NRC staff and the 1.2 multiplier is a reasonable enhancement for conservatism; therefore, the NRC staff finds the proposed GOTHIC methodology for the containment temperature verification acceptable.

3.4.3 Application 8: Equipment Qualification (EQ) Temperature Validation

The licensees' proposed methodology for EQ temperature validation consists of adding a small conductor for the equipment in the containment response GOTHIC model with the appropriate break scenario and single failure consideration that fits the particular equipment's characterstics. The condensation option for the direct heat transfer package is set to Uchida with a constant multiplier of 4.0, consistent with NUREG-0588. Both the natural and forced convection heat transfer options are activated. The convective heat transfer coefficient is calculated using the blowdown rate and the containment free volume, consistent with NUREG-0588. A characteristic length appropriate for the particular equipment is input.

The proposed methodology is consistent with the NRC staff's guidance in NUREG-0588; therefore, the NRC staff finds the proposed GOTHIC methodology for EQ temperature validation acceptable.

3.4.4 Application 9: Transient performance of closed cooling loops for heat exchangers associated with the ECCS and containment heat removal systems.

GOTHIC heat exchanger component modeling has been previously reviewed and approved by the NRC staff as part of the GOTHIC methodology for containment response to LOCA and MSLB events. The proposed methodology for transient performance of closed cooling loops for heat exchangers associated with the ECCS and containment heat removal systems is an incremental change to the LOCA and MSLB peak containment pressure and temperature analyses; therefore, this is acceptable to the NRC staff.

4.0 <u>CONCLUSION</u>

The NRC staff finds the licensees's GOTHIC computer code methodologies, as documented in Topical Report DOM-NAF-3, acceptable subject to the following conditions: (1) Prior to the implementation of the GOTHIC post-reflood mass and energy methodology contained in this topical report for North Anna 1 and 2, Millstone 2 and 3, and Kewaunee, the licensees shall perform bench marking similar to the one performed for Surry 1 and 2 to ensure conservative values are calculated; (2) The GOTHIC NPSHA methodology contained in this topical report cannot be used for other plants that do not credit containment overpressure to calculate NPSHA in their licensing bases.

The NRC staff concludes that sufficient conservatism has been incorporated in the licensees' methodologies to provide assurance that adequate margins to design values will be maintained to satisfy regulatory requirements.

Principal Contributor: G. Tesfaye

Date: August 20, 2006

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