September 10, 2001

Mr. Mark Reddemann
Site Vice President
Kewaunee and Point Beach Nuclear Power Plants
Nuclear Management Company, LLC
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT - REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO. MB0306)

Dear Mr. Reddemann:

The Nuclear Regulatory Commission (NRC) staff has reviewed the topical report WPSRSEM-NP, Revision 3 submitted by Nuclear Management Company (NMC), LLC in a letter dated October 12, 2000, and amended by letters dated February 7, March 7, April 13, and July 26, 2001. The report describes NMC's reload safety evaluation methods for the Kewaunee Nuclear Power Plant. The existing version, Revision 2, of the report WPSRSEM-NP was approved by the NRC staff in 1988. The current submittal, Revision 3, provides an update to reflect methodology changes including the use of RETRAN-3D computer code in the two-dimensional (2D) mode for system responses, the Westinghouse loss-of-coolant accident (LOCA) methodologies for LOCA analysis, and a new critical heat flux correlation for thermal-hydraulic analysis. Based on our review, the NRC staff concludes that Revision 3 is acceptable.

However, the benchmark analyses for the plant-specific applications of RETRAN-3D used in the 2D mode are not performed for the uncontrolled rod withdrawal from a subcritical condition, startup of an inactive coolant loop, anticipated transient without scram, main steamline break (MSLB) and control rod ejection events. You are required to provide the analysis of the five events for the NRC staff to review and approve prior to using RETRAN-3D in the 2D mode in licensing analyses for these events.

The proposed reload methods discussed in WPSRSEM-NP Revision 3 include the use of the CONTEMPT and GOTHIC containment thermal-hydraulic analysis codes. The licensee has in the past, and may in the future, also depend on the COCO containment analysis code. This is a Westinghouse code and calculations using this code would be performed by Westinghouse. It is therefore not included in WPSRSEM-NP Revision 3 and is not evaluated in the enclosed safety evaluation (SE). The licensee's use of the CONTEMPT code has previously been approved by the NRC staff. Therefore, only the use of the GOTHIC code is evaluated in the attached SE.

The NRC staff finds the proposed methods to be acceptable for analysis of the design-basis LOCA and MSLB when used as described in the licensee's April 13, 2001, and July 26, 2001, letters.

The licensee stated (page 36 of Attachment 1 of the April 13, 2001 letter) that there are no other licensing basis uses of the GOTHIC code besides LOCA and MSLB analyses.

The licensee has stated that the Kewaunee GOTHIC containment evaluation model will not be used to calculate the minimum pressure for LOCA backpressure analyses required for demonstrating compliance with the LOCA criteria of 10 CFR 50.46. Therefore, the Kewaunee GOTHIC LOCA containment evaluation model is not approved for these calculations.

The licensee stated that the Kewaunee GOTHIC containment evaluation model will not be applied to subcompartment analyses. Therefore, the Kewaunee GOTHIC LOCA containment evaluation model is not approved for these calculations.

The NRC staff also finds the entrainment model used for the MSLB calculations to be acceptable. This resolves an unreviewed safety question raised during a previous review.

If you have any questions regarding this matter, please contact me at (301) 415-1446.

Sincerely,

/RA/

John G. Lamb, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-305

cc w/encl: See next page

The licensee stated (page 36 of Attachment 1 of the April 13, 2001 letter) that there are no other licensing basis uses of the GOTHIC code besides LOCA and MSLB analyses.

The licensee has stated that the Kewaunee GOTHIC containment evaluation model will not be used to calculate the minimum pressure for LOCA backpressure analyses required for demonstrating compliance with the LOCA criteria of 10 CFR 50.46. Therefore, the Kewaunee GOTHIC LOCA containment evaluation model is not approved for these calculations.

The licensee stated that the Kewaunee GOTHIC containment evaluation model will not be applied to subcompartment analyses. Therefore, the Kewaunee GOTHIC LOCA containment evaluation model is not approved for these calculations.

The NRC staff also finds the entrainment model used for the MSLB calculations to be acceptable. This resolves an unreviewed safety question raised during a previous review.

If you have any questions regarding this matter, please contact me at (301) 415-1446.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WPSRSEM-NP, REVISION 3

RELOAD SAFETY EVALUATION METHODS

NUCLEAR MANAGEMENT COMPANY, LLC.

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION FOR RETRAN

By letter dated October 12, 2000 (Ref. 1), as supplemented by letters dated February 7, March 7, April 13, and July 26, 2001 (Refs. 2 and 3), Nuclear Management Company (NMC), LLC (the licensee) submitted a topical report, WPSRSEM-NP, Revision 3, "Reload Safety Evaluation Methods for Application to Kewaunee," for the Nuclear Regulatory Commission (NRC) staff to review and approve for the Kewaunee Nuclear Power Plant (KNPP).

The topical report, WPSRSEM-NP, describes the licensee's reload analysis methodologies for the KNPP. It contains information related to general physics methods and safety evaluation (SE) methods for loss-of-coolant accident (LOCA) and non-LOCA transient analyses. The licensee uses the reload analysis methods for safety analyses to ensure that the Kewaunee reactor with reload cores can be operated to a specific power level for a specific number of days within the acceptable safety criteria.

The topical report (WPSRSEM-NP) describes the calculation of the following safety parameters:

(1) moderator temperature reactivity coefficient,
(2) power reactivity coefficient,
(3) Doppler reactivity coefficient,
(4) boron reactivity coefficient,
(5) shutdown margin,
(6) scram reactivity curve,
(7) nuclear heat flux hot channel factor,
(8) nuclear enthalpy rise hot channel factor,
(9) effective delayed neutron fraction,
(10) prompt neutron lifetime,
(11) fuel temperature,
(12) maximum assembly average peaking factor,
(13) axial offset at 100 percent power,
(14) maximum core average power in low power assemblies,
(15) maximum 95/95 power for the hot rod,

(16) critical boron concentration, and
(17) fuel rod census.

For each reload application, the licensee reviews the reference analyses of all LOCAs and non-LOCA transients. In the review, the licensee evaluates the effects of plant control parameters, fuel, neutronic and thermal-hydraulic parameters, and engineering safety features on plant transients and accidents. For the cases that are bounded by the corresponding cases in reference calculations, the licensee determines that a re-analysis of the transients is not needed. For cases that are more limiting than the corresponding reference cases, the licensee performs a re-analysis of the affected cases using the NRC-approved methods described in report WPSRSEM-NP for the safety non-LOCA and LOCA transient analysis.

The WPSRSEM-NP report addresses the following LOCA and non-LOCA transients which are considered in the reload analysis:

(1) uncontrolled rod cluster control assembly (RCCA) withdrawal from a sub-critical condition,
(2) uncontrolled RCCA withdrawal at power,
(3) control rod misalignment,
(4) control rod drop,
(5) chemical and volume control system malfunction,
(6) startup of an inactive reactor coolant loop,
(7) excessive heat removal due to feedwater system malfunction,
(8) excessive load increase,
(9) loss of external load,
(10) loss of normal feedwater flow,
(11) loss of reactor coolant flow - pump trip,
(12) loss of reactor coolant flow - locked rotor,
(13) fuel handling accident,
(14) main steamline break (MSLB),
(15) RCCA ejection,
(16) LOCA accident, and
(17) power distribution control verification.

The current version of the WPSRSEM-NP (Revision 2) report was reviewed and approved by the NRC staff in 1988 (Refs. 4 and 5).

Revision 3 of WPSRSEM-NP (Attachment 2 of Ref. 1) is proposed by the licensee to supersede Revision 2 for reload analyses applicable to all Kewaunee future reload cycles after and including Cycle 25.

2.0 EVALUATION OF RETRAN

The proposed Revision 3 of the WPSRSEM-NP retains essentially the same information described in Revision 2 with updated information reflecting methods changes used for safety analyses. Revision 3 incorporates the following changes:

1) Use of RETRAN-3D in the two dimensional (2D) mode to calculate the system responses for non-LOCA transients,

2) Use of the VIPRE-01 code with the high thermal performance (HTP) critical heat flux (CHF) correlation to calculate the departure from nucleate boiling ratios (DNBRs) for fuels with the HTP grid/spacer design provided by Siemens Power Corporation (SPC),

3) Use of referencing Westinghouse LOCA methodologies for small- and large-break LOCA analyses and,

4) editorial changes, including corrections to the limiting directions of core physics parameters and clarification of the definitions of core physics parameters.

The NRC staff has reviewed the proposed changes (Refs. 1 through 3) to the reload analysis methodologies relating to the use of RETRAN-3D, the HTP CHF correlation, LOCA analysis methods and editorial changes listed in items 1 through 4 above. The following is the NRC staff evaluation.

2.1 <u>Non-LOCA Transient Analysis Methodology</u>

Topical report WPSRSEM-NP (Appendixes B through E of Refs. 1 and 4) describes methodologies used for the non-LOCA transient analysis. The transient analysis uses the following computer codes:

DYNODE-P: This code provides a simulation of the system response and calculates system parameters such as core power, reactor coolant system (RCS) flow, primary and secondary side temperatures and pressures during a non-LOCA transient.

VIPRE-01: The VIPRE-01 code provides a simulation of the hot channel thermal hydraulic analysis and determines the minimum DNBRs using the approved CHF correlations.

TOODEE-2: This code provides a simulation of the hot fuel rod and associated coolant channel and computes the transient temperature response for certain accidents. TOODEE-2 is used when VIPRE-01 hot channel yields a DNBR less than the safety DNBR limit.

RETRAN-3D: An application of RETRAN-3D limited to the two-dimensional (2D) mode is also used to simulate the system response for transient analyses. The licensee's use of RETRAN-3D does not include any of the non-equilibrium or three-dimensional (3-D) core modeling techniques.

The NRC staff finds that DYNODE-P, VIPRE-01 and TOODEE-2 were previously reviewed and approved (Reference 5) by the NRC staff for use in the analysis for KNPP licensing applications. The NRC staff also finds that the proposed use of these codes described in Revision 3 of WPSRSEM-NP is within the applicable ranges of the approved codes. Therefore, the NRC staff concludes that these codes continue to be acceptable for referencing in reload licensing applications.

The use of RETRAN-3D in the 2D mode is proposed by the licensee to replace RETRAN-02 for calculating the system response during a non-LOCA transient. The NRC staff has reviewed the licensee's supporting analysis (Ref. 1) and its response to the NRC staff's request for additional information (Refs. 2 and 3) for acceptability of the proposed RETRAN-3D methods and prepared the following evaluation:

2.1.1 Acceptability of Use of RETRAN-3D in the Two-Dimensional Mode for Calculating Transient System Responses

RETRAN is a thermal-hydraulic computer code that is used to evaluate the effects of various upset reactor conditions in the RCS. WPSRSEM-NP-NP-A, Revision 2 documented the use of RETRAN-02, which is an earlier version of RETRAN. RETRAN-02 was used by the licensee to verify analysis using DYNODE-P or to independently perform a transient analysis.

Revision 3 of WPSRSEM-NP proposes use of RETRAN-3D in the 2D mode, replacing RETRAN-02, for performing the non-LOCA transient analysis. While functionally equivalent to RETRAN-02, RETRAN-3D is the most recent version of the RETRAN code. The RETRAN-3D code incorporates new models and equations, including additional balance equations to predict non-equilibrium phenomena and three-dimensional (3D) core kinetics, as well as advanced numerical solution methods and new correlations. RETRAN-3D was recently reviewed and approved by NRC for licensing applications (Ref. 6.) The NRC staff's generic approval of RETRAN-3D is subject to a number of limitations described in the SE. During the same RETRAN-3D review, the NRC staff also determined that the use of RETRAN-3D in the 2D mode is acceptable. However, the NRC staff SE (Ref. 6) requires that when the RETRAN-3D code is used in the 2D mode for safety analyses, any of the following new RETRAN-3D models cannot be used:

(1) generalized laminar friction model,
(2) dynamic gap conductance model,
(3) accumulator model,
(4) dynamic flow regime model,
(5) new control blocks added to improve functionality,
(6) Govier horizontal flow regime map and stratified flow friction model,
(7) Chexal-Lellouche drift flux model,
(8) method of characteristics enthalpy option,
(9) noncondensable gas flow model,
(10) three-dimensional core kinetics, and,
(11) the five-equation nonequilibrium model,

In response to the NRC staff's request, the licensee has evaluated (Refs. 2 and 3) its compliance with the conditions specified in the SE (Ref. 6) for RETRAN-3D and confirmed that the SE conditions for generic approval of RETRAN-3D are met, and none of the new RETRAN-3D models listed above is included in RETRAN-3D used in the 2D mode. Accordingly, the NRC staff concludes that the licensee adequately addresses the NRC staff concerns relating to conformance to SE conditions.

To support the adequacy of the use of RETRAN-3D in the 2D mode for the KNPP plant specific applications, the licensee performed benchmark analyses comparing the licensee safety analysis computer codes, RETRAN-3D in the 2D mode and DYNODE-P. The benchmark analyses were performed for Kewaunee design-basis non-LOCA transients including:

(1) uncontrolled RCCA withdrawal at power,
(2) chemical and volume control system malfunction,
(3) excessive heat removal due to feedwater system malfunction,
(4) excessive load increase,
(5) loss of external electrical load,
(6) loss of normal feedwater, and
(7) locked rotor.

For each case analyzed, the licensee used its current non-LOCA transient analysis methodologies with similar analysis inputs (such as geometry, power level, fluid conditions etc.) and subsystem models (such as charging/letdown, steam generators etc.) assumptions. The model-assumptions results of benchmark analyses (in Attachment 3 of Ref. 1) show that there are no unexpected results and the calculational results with RETRAN-3D in the 2D mode and DYNODE-P for the cases analyzed are in good agreement.

Since the licensee has (1) satisfactorily addressed the SE conditions for use of the RETRAN-3D code, (2) showed no unexpected calculational results, and (3) demonstated good agreement between the RETRAN-3D results and the DYNODE-P results, the NRC staff concludes that RETRAN-3D used in the 2D mode is acceptable for non-LOCA transient analyses. However, the benchmark analyses for the plant-specific applications of RETRAN-3D used in the 2D mode are not performed for the uncontrolled rod withdrawal from subcritical condition, startup of an inactive coolant loop, anticipated transient without scram, main steamline break and control rod ejection events. The licensee is required to provide the analysis of the five events for the NRC staff to review and approve prior to using RETRAN-3D in the 2D mode in licensing analyses for these events.

2.2 <u>DNBR Calculations Using the VIPRE-01 Code with the HTP CHF Correlation</u>

The thermal-hydraulic analyses are performed by the licensee to establish the maximum allowable power distribution limits to maintain the required margin to DNB at various coolant flows, temperatures and pressures. Appendix C of WPSRSEM-NP describes the methods of thermal-hydraulic analysis. The VIPRE-01 code is used for a simulation of the hot channel thermal-hydraulic analysis and determines the minimum DNBR using the approved CHF correlations. With use of VIPRE-01 and the approved CHF correlations, the safety DNBR limits are established to provide 95 percent probability of precluding DNB, and thus, avoiding fuel failures at a 95 percent confidence level.

WPSRSEM-NP-A, Revision 2 (Ref. 4) referenced the Westinghouse W-3 CHF correlation with a NRC-approved safety DNBR limit for DNBR calculations for all the non-LOCA transient an NRC analyses, including the steamline break (SLB) analysis. The use of Westinghouse W-3 CHF correlation, with associated safety DNBR limit was approved by the NRC staff for the licensee to apply to the fuel design in use at Kewaunee at that time.

Revision 3 of WPSRSEM-NP (Appendix C of Ref. 1) removes the Westinghouse W-3 correlation and replaces it with the HTP CHF correlation with a safety DNBR limit of 1.14 for all transients except the main steamline break. For the current operating cycle, Cycle 24, and future cycles, the licensee uses SPC fuels that incorporate the HTP grid/spacer design. The HTP grid/spacer fuel has a corresponding CHF correlation and safety DNBR limit. The NRC staff finds that the safety DNBR limit calculated with the VIPRE-01 code and HTP CHF correlation for the HTP grid/spacer design was reviewed and approved by NRC staff in December 1997 (Ref. 7) for KNPP reload applications. For the HTP fuel design, the SLB analysis results in thermal-hydraulic conditions outside the applicable range of the HTP CHF correlation. The licensee proposes in Revision 3 of WPSRSEM-NP that the Westinghouse W-3 correlation with a previously approved safety DNBR limit be used for the SLB thermal analysis. Since (1) the Westinghouse W-3 correlation was previously used by the licensee to calculate DNBRs during an SLB event for fuels with the SPC Bi-Metallic grid design, and (2) the SPC HTP grids/spacers allow better mixing of coolant within a fuel assembly and improve the DNBR performance, the NRC staff concludes that the use of Westinghouse W-3 correlation is conservative and is acceptable.

2.3 Methods for the Small- and Large-Break LOCA Analysis

The licensee evaluated LOCA events to ensure that the analytical results meet the 10 CFR 50.46 requirements with respect to the peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long-term cooling. Sections 2.12 through 2.15 and 3.16 of Revision 3 of WPSRSEM-NP contain information relating to the LOCA analysis. The licensee indicates that the NOTRUMP code is used for small-break LOCA analysis. The NOTRUMP code consists of the modeling features that meet the requirements of Appendix K to 10 CFR Part 50. As documented in WCAP-10054-A (Ref. 8), the NRC staff previously approved the NOTRUMP code for the small-break LOCA analysis.

The methods for large-break LOCA (LBLOCA) analysis referenced in Revision 3 of WPSRSEM-NP are changed to reflect the licensee’s implementation of upper plenum injection (UPI) safety analysis methods for Kewaunee. The LBLOCA UPI safety analyses (the actual system analysis of the LBLOCA accident) are performed by Westinghouse using Westinghouse methods. WPSRSEM-NP-A, Revision 2, did not include LBLOCA safety analysis methods since at that time the analysis of record for LBLOCA was a non-UPI LOCA methodology.

The licensee’s large-break LOCA analysis methods are described in the Westinghouse topical report WCAP-10924, “Westinghouse Large-Break LOCA Best-Estimate Methodology.” This LBLOCA methodology uses the approach described in SECY 83-472, “Emergency Core Cooling System Analysis Methods.” The SECY 83-472 approach allows for a substantial amount of the conservatism contained in typical emergency core cooling system (ECCS) evaluation models to be reduced in a systematic manner while still complying with the requirements set forth in Appendix to 10 CFR Part 50. As documented in WCAP-10924-A (Ref. 9) and an NRC letter (Ref. 10), the NRC staff previously approved the UPI LOCA analysis methods for the KNPP LBLOCA analysis.

Since the NRC staff finds both small and LBLOCA analysis methodologies were previously approved by the NRC staff, the NRC staff determines that the licensee’s approach to reference the NRC-approved methodologies for LOCA analyses does not invalidate the acceptance of WPSRSEM-NP and is acceptable.

2.4 Editorial Changes

The NRC staff has reviewed additional changes to Revision 2 of WPSRSEM-NP discussed in Attachments 1 and 2 of Reference 1. The changes fall into the following categories:

(1) corrections to the limiting direction of the effective delayed neutron fraction,

(2) clarification of the definitions of core physics parameters, including critical boron concentration, boron reactivity coefficient, scram reactivity curves, fuel rod census and nuclear enthalpy rise hot channel factor,

(3) continued use of the least negative values for the Doppler and moderator temperature coefficients in the analysis of the control rod ejection event,

(4) deletion of detailed discussions of sensitivity studies, which were used to support original approval of the VIPRE-01 code described in Appendix C of WPSRSEM-NP,

(5) deletion of a discussion of best estimate safety analysis methods in Appendix F, which is no longer applicable in the topical report WPSRSEM-NP, and

(6) removal of results of transient analyses from Section 3.0 to Appendix G of WPSRSEM-NP.

The proposed changes listed in categories 1 and 2 affected the following sections of the WPSRSEM-NP report:

·Section 3.1.5 - Uncontrolled Rod Withdrawal from a Sub-critical Condition
·Section 3.2.5 - Uncontrolled Rod Withdrawal at Power
·Section 3.5.5 - Chemical and Volume Control System Malfunction
·Section 3.7.5 - Excessive Heat Removal due to Feedwater System Malfunction
·Section 3.8.5 - Excessive Load Increase
·Section 3.12.5 - Loss of Reactor coolant Flow - Locked Rotor Coolant
·Section 3.13.5 - Fuel Handling Accident

The NRC staff finds that the proposed changes of the six categories discussed in this section are editorial in nature and do not reduce conservatism of the methods discussed in the WPSRSEM-NP. Therefore, the NRC staff concludes that these changes are acceptable.

3.0 CONCLUSION FOR RETRAN

The licensee is using the reload methodologies described in report WPSRSEM-NP to ensure that the Kewaunee reactor with reload cores can be operated to a specific power level for a specific number of days within the acceptable safety criteria. Revision 2 of WPSRSEM-NP was previously approved by NRC staff. Revision 3 provides an update to reflect method changes relating to use of RETRAN-3D in the 2D mode for non-LOCA transient analyses, the Westinghouse LOCA methodologies for LOCA analyses and the HTP CHF correlation for thermal-hydraulic analyses. Since the NRC staff finds that these method changes are acceptable for use in the Kewaunee reload analyses, the NRC staff concludes that Revision 3 of WPSRSEM-NP is acceptable. Our approval of Revision 3 does not remove or change the limitations stated in the NRC's safety evaluation reports (SERs) for the topical reports referenced in Revision 3 of WPSRSEM-NP.

However, the benchmark analyses for the plant-specific applications of RETRAN-3D used in the 2D mode are not performed for the uncontrolled rod withdrawal from a subcritical condition, startup of an inactive coolant loop, anticipated transient without scram, main steamline break and control rod ejection events. The licensee is required to provide the analysis of the five events for the staff to review and approve prior to using RETRAN-3D in the 2D mode in licensing analyses for these events.

4.0 INTRODUCTION FOR GOTHIC

An October 12, 2000, letter from NMC proposed performing reload analyses for the KNPP in accordance with Revision 3 to WPSRSEM-NP. The original October 12, 2000, letter was supplemented by letters dated April 13, 2001, and July 26, 2001. Included in this revision is a description of proposed analysis methods for performing containment thermal-hydraulic calculations for the postulated design-basis LOCA and the postulated design-basis MSLB.

The proposed reload methods discussed in WPSRSEM-NP Revision 3 encompass the use of the CONTEMPT[1] and GOTHIC[2] containment thermal-hydraulic analysis codes. CONTEMPT is an NRC-developed code. The licensee's October 12, 2000, letter states that CONTEMPT has been applied to containment analysis for Kewaunee since original licensing and references several instances where it has been applied since then. GOTHIC is developed and maintained for the Electric Power Research Institute (EPRI) by Numerical Applications Inc. Use of GOTHIC by the licensee has not been previously approved by the NRC. The licensee has in the past, and may in the future, also depend on the COCO[3] containment analysis code. COCO is a Westinghouse code and calculations using this code would be performed for the licensee by Westinghouse. Since the licensee would not perform calculations using COCO, COCO is not included in WPSRSEM-NP-NP Revision 3. Therefore, since GOTHIC is the only code which has not been previously approved for Kewaunee, only the use of GOTHIC is evaluated in this safety evaluation.

5.0 BACKGROUND FOR GOTHIC

The licensee's October 12, 2000, letter to the NRC transmitted Revision 3 to WPSRSEM-NP for NRC review and approval. WPSRSEM-NP Revision 2 did not include containment thermal-hydraulic analysis methods. Revision 3 to WPSRSEM-NP adds the containment thermal-hydraulic analysis methods applied to the design-basis LOCA and the design-basis MSLB for the KNPP.

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[1] Don W. Hargroves and Lawrence J .Metcalf, CONTEMPT-LT/028-A A Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," NUREG/CR-0255 March 1979.

[2] George, Thomas L., "GOTHIC Version 6.0, Containment Analysis Package," December 1997, EPRI RP4444-1.

[3] F. M. Bordelon and E. T. Murphy, "Containment Pressure Analysis Code (COCO)," WCAP-8327 (Proprietary), WCAP-8326 (Non-Proprietary) Westinghouse Electric Corporation, July 1974 (Proprietary).

Kewaunee is currently in Cycle 24. For Cycle 25, the licensee's July 26, 2001, letter states that Westinghouse will provide the containment LOCA analysis using COCO and the licensee will perform any additional containment LOCA analyses using CONTEMPT. For future cycles (beyond Cycle 25), the containment LOCA analysis will either be performed by a vendor (e.g., Westinghouse) using the vendor's NRC-approved containment methods or by the licensee using GOTHIC or CONTEMPT. The mass and energy release calculations which are an input to the containment analysis will be performed by the LOCA analysis vendor. The licensee will not perform mass and energy release calculations for the LOCA.

For Cycle 25, CONTEMPT will be used by the licensee for the MSLB containment thermal-hydraulic analyses. For future cycles (beyond Cycle 25), the containment analysis will be performed by the licensee using GOTHIC or CONTEMPT. The licensee will also perform the mass and energy release calculations for the MSLB. This is discussed later in this safety evaluation.

The licensee has benchmarked the GOTHIC code against analyses performed with CONTEMPT and COCO. This is documented in WCAP-15427 Revision 1[4]. Therefore, the review of WCAP 15427 Revision 1 is also addressed in this SER. In addition, the GOTHIC code documentation describes validation of the GOTHIC code with available data and analytical solutions. Benchmarking is discussed further in Sections 3.2.7 and 3.5.5 of this safety evaluation.

The licensee's October 12, 2000, letter did not provide a description of the proposed GOTHIC evaluation model. The licensee's July 26, 2001, letter proposes that the GOTHIC model described in WCAP 15427 Revision 1 be the evaluation model for the KNPP. The licensee provided more detail on this evaluation model and provided justification for this evaluation model in addition to that in WCAP 15427 Revision 1 in letters dated April 13, and July 26, 2001. Thus, the model used in WCAP 15427 Revision 1 and described in more detail in these letters constitutes the Kewaunee GOTHIC evaluation model.

The licensee's July 26, 2001, letter states that:

> All evaluation models for reload safety evaluation methods, including the GOTHIC EM [evaluation model], are developed and maintained as per applicable quality assurance programs consistent with 10 CFR 50, Appendix B and GL [NRC Generic Letter] 83-11, Supplement 1[5]. In conformance with the above references, changes to an evaluation model that represent a departure from approved methods will require NRC approval.

The final calculations of the responses of the KNPP to the LOCA and the MSLB have not been calculated with the new evaluation model (EM). Only sensitivity studies, described in

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[4]R. Ofstun, "Development and Qualification of a GOTHIC Containment Evaluation Model for the Kewaunee Nuclear Power Plant," WCAP 156427 Revision 1 (Proprietary). WCAP 15667 (April 2001) is a non-proprietary version of the same document. This SER will reference only the proprietary version.

[5]GL 83-11 Supplement 1, "Licensee Qualification for Performing Safety Analyses," June 24, 1999.

WCAP 15427 Revision 1 and the licensee's letters dated April 13, and July 26, 2001, have been provided to the NRC. This is acceptable since the models and assumptions have been adequately defined and the degree of conservatism to be included in the important input parameters has been described. Variations in input parameters within the bounds set by this review will not change the conclusions of this review.

6.0 EVALUATION OF GOTHIC

6.1 Initial Conditions

The initial boundary conditions for containment analyses of the LOCA and MSLB were provided by the licensee in Table 1 of the April 13, 2001, letter and revised in the licensee's July 26, 2001, letter. The initial containment pressure, temperature and relative humidity have values of 16.85 pounds per square inch absolute (psia), 120° Fahrenheit (F), and 17.7 percent, respectively. Section 3.6.d of the Kewaunee technical specifications (TSs) states that the pressure should not exceed 2 psi. The higher initial pressure is therefore conservative with respect to this TS. The assumption of 120 °F is typically used as a conservative initial containment temperature. A low value of relative humidity is conservative. The licensee's assumption of 17.7 percent relative humidity is consistent with previously approved analyses for Kewaunee.

The licensee assumes an initial service water temperature of 80 °F. This value is not specified in the TSs. However, in response to a NRC staff question, the licensee states that:

> There is an alarm in the control room on service water temperature that alerts the operator to a high (approximately 76 °F) service water temperature. The alarm and the associated alarm response actions help to assure that the KNPP [Kewaunee Nuclear Power Plant] SWS [Service Water System] temperature will not exceed the 80 °F value.
>
> A historical review of service water inlet temperature dating back to 1990 was performed and at no time in that period did the temperature exceed 80 °F.

The net free volume of the containment is given in Table 1 as 1.32×10^6 cubic feet (ft^3). The licensee states that this is a lower bound value. This is conservative for peak pressure calculations.

The proposed containment analysis methods do not take credit for sprays during recirculation. Therefore, the only active heat removal credited during recirculation is by the fan cooler units (FCUs). The licensee stated that conservatively low containment FCU capability is assumed in the containment integrity analysis for the design-basis accidents. The conservative assumptions are discussed in the licensee's July 26, 2001, letter. The fouling factor is chosen conservatively based on performance monitoring results. The NRC staff finds the licensee's modeling of the containment FCUs to be acceptable.

The licensee has assumed worst single failures for the LOCA and MSLB analyses. The licensee stated in the April 13, 2001, letter, that a series of analyses were done using different break sizes and locations for the containment LOCA response.

In the April 13, 2001, letter, the licensee stated that both the current containment design-basis accident (DBA) evaluation models (COCO and CONTEMPT) and the proposed, benchmarked GOTHIC containment DBA evaluation model represent the containment as a single lumped parameter volume. The use of a single volume model is, in general, conservative and therefore the NRC staff finds this acceptable.

The licensee's April 13, 2001, letter, stated that a lower bound estimate on the number and surface area of the structural heat sinks is used in the analysis. This is conservative for peak pressure and temperature calculations.

6.2 <u>Containment LOCA Thermal Hydraulic Analyses</u>

6.2.1 LOCA Mass and Energy

An important consideration in containment peak pressure and temperature calculations is the mass and energy of the reactor coolant expelled from the RCS to the containment atmosphere. The mass and energy releases for the minimum and maximum safeguards cases for the double ended pump suction (DEPS) and the releases from the blowdown of a double ended hot leg (DEHL) break are calculated by Westinghouse. A brief description of this calculation is given in the KNPP updated safety analysis report (USAR). The Westinghouse SATAN computer code is used for this calculation. The USAR states that:

> the energy transferred from the core to the coolant for the containment evaluation far exceeds that transferred from the core thermal evaluation...a conservatively high core heat transfer coefficient is used for the containment evaluation...

6.2.2 LOCA Single-Failure Assumptions

As part of the LOCA evaluation, both the DEPS and the DEHL breaks are analyzed to determine the worst break location. Due to the early peak pressure and temperature of the DEHL case, no single failures are considered since containment engineered safeguards equipment would not actuate in time to affect the peak conditions.

The GOTHIC evaluation model considers both a maximum safeguards and a minimum safeguards case for the DEPS.

For the maximum safeguards DEPS case (defined by the Kewaunee licensee as the failure of anything other than an emergency diesel generator [EDG]), a failure of a containment spray pump (CSP) was assumed as the worst single-failure. This leaves one CSP and four containment FCUs available as active heat removal systems. The licensee also considered a minimum safeguards case (defined by the Kewaunee licensee as the loss of an EDG). This leaves one CSP and two containment FCUs available for active heat removal.

The licensee has done a thorough single-failure evaluation. The NRC staff finds the licensee's single-failure evaluation to be acceptable.

6.2.3 Modeling of Mass and Energy Release in the Containment Atmosphere

In containment thermal-hydraulic calculations the mass and energy of the reactor blowdown are assumed to be distributed into the vapor and liquid regions of the containment. The assumptions made regarding these distributions influence the containment pressure and temperature.

In the July 26, 2001, letter, the licensee stated that:

> The Kewaunee GOTHIC containment evaluation model forces the blowdown break flow to be released to the containment atmosphere as drops. The drops initially flash to steam at the saturation temperature corresponding to the local containment pressure, then continue to evaporate as they come into equilibrium with the temperature of the atmosphere.

WCAP 14527 Revision 1 provides a sensitivity study that shows that increasing the break flow drop size used by GOTHIC by a factor of ten provides better agreement between the peak containment temperature calculated by GOTHIC and the higher temperature calculated by CONTEMPT. The licensee stated in the July 26, 2001, letter, that the drop size used in the GOTHIC calculation is based on recommendations in the GOTHIC Users Manual. This value, in turn, is based on experiments reported in the American Chemical Engineering Journal[6]. The licensee reviewed these data and concluded that the break flow drop size assumption used in GOTHIC (100 microns) is greater (therefore more conservative) than the values given in the paper. The NRC staff therefore finds the licensee's use of the recommended GOTHIC break flow drop size to be acceptable.

6.2.4 Heat Transfer Coefficients

The containment structure is an important heat sink early in the LOCA or MSLB. The heat transfer to the containment structure depends on the heat transfer coefficient.

The licensee's April 13, 2001, letter, states that all three evaluation models (COCO, CONTEMPT and GOTHIC) use the Tagami[7] correlation for condensation heat and mass transfer during blowdown and the Uchida[8] correlation for post-blowdown heat transfer.

The Tagami and Uchida correlations give conservative results for condensation heat transfer in air and are acceptable as applied in the topical report.

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[6] R. Brown and J. L. York, "Sprays Formed by Flashing Liquid Jets," AIChE Journal Volume 8, No.2, May 1962.

[7] T. Tagami, "Interim Report on Safety Assessment and Facilities Establishment Project in Japan for Period Ending June 1965 (No. 1)," prepared for the National Reactor Testing Station, February 28, 1966.

[8] H. Uchida, A. Oyama, and Y. Toga, "Evaluation of Post-Incident Cooling Systems of Light-Water Power Reactors," Proc. Third International Conference on Peaceful Uses of Atomic Energy, Volume 13, Session 3.9, United Nations, Geneva (1964).

For concrete surfaces, 40 percent of the heat transfer coefficient calculated for steel surfaces is used. This is in accordance with standard practice and is acceptable.

6.2.5 Spray Modeling

One of the most effective means to reduce containment pressure (following the blowdown pressure peak) is by the internal spray. The spray water emerges from the spray nozzles as small droplets giving a very large surface area. Experiments have shown that spray water attains the same temperature as the external medium (vapor) within one or two feet of the spray nozzle. WCAP 8326[9], the non-proprietary version of the COCO topical report, provides several references supporting this.

GOTHIC has the capability to model heat and mass transfer from spray droplets. The licensee's April 13, 2001, letter, states that a small constant diameter spray drop is input to the DBA evaluation model. This effectively simulates the assumption in COCO of instantaneous equilibration of the spray droplet with the vapor. It also is consistent with CONTEMPT which assumes 100 percent spray efficiency. The Kewaunee USAR, Section 14.3.4, "Containment Integrity Evaluation," provides justification for a spray efficiency of 100 percent.

The licensee further justified this assumption in response to a NRC staff request for additional information in the July 26, 2001, letter. The licensee stated that:

> The results of spray tests by Parsly at Oak Ridge National Labs (ORNL-TM-2412 Part VI, January 1970) justify the use of 100 percent spray efficiency for a full height dry containment building [such as Kewaunnee's]. These tests verify that the temperature of a spray droplet falling through the air/steam mixture would equilibrate with the atmosphere within a relatively short fall height. Therefore, the actual spray efficiency will be essentially 100 percent.
>
> Although using a spray efficiency input value of less than 100 percent would be conservative, other conservative assumptions in the containment spray system modeling ensure a conservative model. The containment spray setpoint, the delay time, and the flow rates are all conservatively assumed. The spray flow is the minimum compared to what the pump can deliver. The containment pressure setpoint is the setpoint plus instrument uncertainties. The spray delay time assumed includes the longest times for the signals to be reached, processing time, diesel start-up, sequencing time, and the time to fill the spray lines.

The NRC staff finds the use of 100 percent spray efficiency to be acceptable .

6.2.6 Interaction of the Containment Atmosphere with the Sump

COCO and CONTEMPT do not allow interfacial heat and mass transfer between the sump and the containment atmosphere. While GOTHIC will model heat and mass transfer between the

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[9] F. M. Bordelon and E. T. Murphy, "Containment Pressure Analysis," WCAP 8326 July 1974.

sump and the containment atmosphere, the Kewaunee evaluation model prevents this interaction by setting the interfacial area to 0.0 ft^2.

6.2.7 GOTHIC Sump Recirculation Modeling

WCAP 15427 Revision 1 describes the modeling of the residual heat removal (RHR) piping and the recirculation spray piping. The sump, the spray pump, the piping volume, the RHR heat exchanger and the component cooling water heat exchanger were modeled. In general, the modeling attempted to simulate the Kewaunee COCO model.

The licensee's April 13, 2001, letter, states that no credit is taken for containment spray during the recirculation phase of the LOCA. This is conservative, but since the spray water source is the containment emergency sump during the recirculation phase, the spray has less effect than during the injection phase when the water source is the refueling water storage tank.

6.2.8 Benchmarking

WCAP 15427 Revision 1 provides comparisons between the GOTHIC code and the COCO code for a Kewuanee DEPS large break LOCA with the worst single-failure being the loss of one EDG at time zero. COCO was used previously to model the Kewaunee containment response to a large break LOCA. For this event, GOTHIC predicted a peak containment pressure of 56.9 psia at about 60 seconds; COCO predicted a peak pressure of 57.7 psia at about 60 seconds. GOTHIC predicted a slightly lower vapor temperature and higher sump temperature after blowdown. The licensee attributes these differences to the fact that GOTHIC has better interfacial heat and mass transfer models and is able to model the two phase flow phase separation better than COCO. This results in more break energy being deposited in the sump with GOTHIC which results in higher sump temperature and lower vapor temperature and pressure. To confirm this explanation, the licensee increased the size of the drop in the break flow by a factor of ten. This reduces the heat transfer from the vapor to the drop and results in a peak containment pressure of 57.6 psia (versus 57.7 psia calculated with COCO). The GOTHIC sump liquid temperature was slightly higher than the original case with the smaller drop size but was greater than the COCO value.

WCAP 15427 Revision 1 also compares the GOTHIC to the COCO evaluation model for a DEHL LOCA. The peak calculated containment pressures were nearly identical (59.5 psia for GOTHIC and 59.4 psia for COCO). Once again, COCO predicts a higher initial vapor temperature and a lower sump temperature.

In addition to these calculations done to compare the proposed and previous evaluation models, the GOTHIC documentation[10] contains a comparison of GOTHIC with a variety of experimental data from different sources as well as with analytical solutions. The NRC has not reviewed and evaluated these comparisons. The conclusion of the EPRI documentation[10] is

> Over a wide range of tests run, GOTHIC predictions compare quite well to calculated or measured parameters.

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[10] GOTHIC Containment Analysis Package Qualification Report Version 4.0 NAI 8907 Rev 2 Prepared for the Electric Power Research Institute September 1993.

Therefore, since, on a best estimate basis GOTHIC compares well with data and analytical solutions and the results of the Kewaunee GOTHIC evaluation model presented in WCAP 15427 Revision 1 and the licensee's April 13, and July 26, 2001, letters, compare well with the COCO evaluation model results for Kewaunee, the NRC staff finds the use of the GOTHIC evaluation model for Kewaunee to be acceptable for calculating the containment response to LOCAs.

6.3 Subcompartment Analyses

The licensee did not present any description of how the GOTHIC code would be applied in performing subcompartment analyses. Therefore, the evaluation model presented in WCAP 15427 Revision 1 and the April 13, and July 26, 2001 letters, is not approved for subcompartment analyses.

6.4 Main Steamline Break

The CONTEMPT LT/28 computer code is used for the current Kewaunee MSLB evaluation model. The licensee proposes to also use the GOTHIC code for the Kewaunee MSLB evaluation model. The GOTHIC LOCA model is modified to calculate the MSLB peak pressure and temperature. Additional heat sinks, described in WCAP 15427 Revision 1 are added to the model. Heat sinks are important to the MSLB evaluation model since the peak pressure occurs before the containment spray and fan coolers become effective. The MSLB evaluation model does not include the GOTHIC models developed for recirculation of sump fluid since these models are not applicable to a MSLB accident.

The revaporization fraction is set to 8 percent. This is in accordance with NUREG-0588[11] and is acceptable.

6.4.1 Initial Conditions

The MSLB peak pressure transient is based on the mass and energy releases from a full double ended rupture of a main steam line (1.4 ft^2), including failure of the feedwater regulating valve, while operating at 102 percent power.

The MSLB peak temperature response is calculated based on the mass and energy release from a 1.1 square foot (ft^2) break in the main steam line, including loss of offsite power and failure of a safeguards train (two FCUs and one CSP) at 0 percent power.

The steam generator (SG) level (i.e., the mass inventory) is an important input to the MSLB calculations. The licensee states that Cycle 25 will be the first cycle with replacement SGs. These SGs will operate with a nominal SG level of 44 percent at all power levels. The initial SG level that is assumed in all of the MSLB containment response safety analyses will be 55 percent at hot zero power and 49 percent at all other power levels. The higher value assumed in the analyses bounds instrumentation errors for the SG level measuring system,

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[11] A. J. Szukiewicz, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," NUREG-0588, Revision 1, July 1981.

including the actual versus the indicated level differences at all power levels. The NRC staff agrees that this is conservative and is acceptable.

6.4.2 Mass and Energy Release for a Main Steamline Break Analysis

Although the licensee will obtain mass and energy release data from a vendor using approved methods for the LOCA analysis, the licensee calculates the mass and energy release for the postulated MSLB.

The licensee's April 13, 2001, letter (Attachment 1 Page 13), provides a brief history of the licensee's development of the capability to perform mass and energy release calculations for the MSLB.

The licensee's April 13, 2001, letter, also provides a description of the calculation of mass and energy release for MSLB calculations. One of the major factors which influence the release of mass and energy following a steamline break is described by the licensee as the "state of the secondary blowdown." This refers, in particular, to the amount of liquid which is included as carryover or entrainment in the steam blowdown from the ruptured SG due to the inability of the SG moisture separators to remove the entrained liquid at certain break sizes and power levels.

By letters dated May 2, and September 25, 1997, the licensee proposed a change to the TSs bases as an unreviewed safety question, requiring NRC review under 10 CFR 50.59. The licensee proposed changing the 5 second main steam isolation valve (MSIV) closure time to 10 seconds in order to allow increased operational flexibility in the SG level. The NRC approved this change in a safety evaluation dated April 15, 1998. However, the NRC staff evaluation stated:

> Due to the significant worth of the revised entrainment assumption, the staff considers the licensee's new mass and energy model to involve an unreviewed safety question. The staff will initiate a separate review of the Kewaunee entrainment model...

The NRC staff has performed this separate review of the entrainment model as part of this review of the GOTHIC evaluation model. The results of this review are given below.

The licensee calculates the entrainment iterating between the DYNODE code (which calculates the system response) and the RETRAN code which models the faulted SG in sufficient detail to give the quality of the SG blowdown as a function of time. Iterations continue between these codes until the entrainment curves have converged to within the acceptance criterion.

Standard Review Plan Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," states that:

> if liquid entrainment is assumed in the steamline break calculations, experimental data should support the predictions of the liquid entrainment model. The effect on the entrained liquid of steam separators located upstream from the break should be taken into account. A spectrum of steamline breaks should be analyzed, beginning with the double-ended break and decreasing in area until no entrainment is calculated to occur, to allow selection of the maximum release case.

Table 6 of the licensee's April 13, 2001, letter, provides a comparison between Combustion Engineering, Inc. experimental entrainment data[12] and RETRAN 3D predictions. The comparisons between the model and data show good agreement. There is no conservatism in these calculations. However, the licensee has proposed a multiplier, based on the comparisons between data and RETRAN 3D predictions, which adds some margin to the entrainment calculations. The multiplier is a 95/95 one-sided upper tolerance limit on the ratio of measured data to predicted values.

The NRC staff asked the licensee to justify why a comparison using RETRAN 3D, with data from an experimental facility which differs in scale and configuration from the Kewaunee SG provides confidence in the ability of RETRAN 3D to predict the Kewaunee SG blowdown behavior following a MSLB. The licensee responded in the April 13, 2001, letter. The licensee proposed that the entrainment phenomenon can be characterized by two ratios: the ratio of the break area to the (steam generator) vessel volume (A_{break}/V_{vessel}) and the ratio of the break area to the (steam generator) vessel free surface area ($A_{break}/A_{free\ surace}$). The first ratio characterizes the depressurization rate and the second ratio characterizes the tendency of the water to separate from the predominantly liquid region and enter the predominantly vapor regions and to flow out the break. The licensee demonstrated that these ratios for the Combustion Engineering test facility and the replacement SGs are nearly in the same range of values. In addition, the qualities in the steam dome of the test facility and the Kewaunee SGs are almost completely within the same range. This demonstrates that the distribution of liquid inventory is comparable between the test facility and the Kewaunee SGs.

Therefore, the NRC staff finds the licensee's proposed use of an entrainment model to be consistent with the guidance of the Standard Review Plan Section 6.2.1.4 and is acceptable.

The quality of the unfaulted SG flow is conservatively assumed to be 1.0.

The licensee uses the 1971 American Nuclear Society decay heat standard with a 20 percent uncertainty to account for the decay of actinides and fission product decay. This standard is conservative and acceptable (Appendix K to 10 CFR 50.46 requires use of this standard for LOCA calculations).

The licensee does not take credit for uncovery of the tubes in the faulted SG. This results in significantly more energy transfer from the RCS than would be expected to occur if tube uncovery was modeled.

6.4.3 MSLB Single-Failure Assumptions

The licensee's April 13, 2001, letter, states that three single failures are considered for the MSLB. These are:

- Failure of one feedwater regulating valve to isolate,
- Failure of one main steam isolation valve (MSIV) to isolate, and

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[12] Kling, C. L. and Peeler, G. B. "Moisture Carryover During An NSSS Steamline Break," CENPD 80, January 1973.

- Failure of one containment safeguards train. This includes failure of one containment spray train and two containment fan cooler units to actuate.

These three failures cover the situations, respectively, of an additional large source of water to the SG which will flash in the feedwater pipe or boil in the SG to become a large additional source of steam; a blowdown of both SGs (with the additional failure of a non return check valve), and reduced mitigating ability. Since these failures cover the possible adverse effects of a failure, the NRC staff finds this choice of single-failure assumptions to be acceptable.

For failure of the feedwater regulating valve, feedwater is isolated by the feedwater isolation valve. The feedwater isolation valve begins to close from full open at a rate of 1.18 percent/sec. The closure of this valve is modeled as an instantaneous closure at the time it would have closed at this closure rate. This is conservative and acceptable.

6.4.4 Heat Transfer Correlations

The Tagami and Uchida correlations are used to calculate heat transfer to the containment structures during a MSLB accident. As discussed in Section 3.2.4 of this SER input, these correlations are conservative and acceptable.

6.4.5 Benchmarking

WCAP 15427 Revision 1 reports that the calculated peak pressure is based on the full double-ended rupture of a main steamline with failure of a feedwater regulating valve from 102 percent power with full safeguards. The peak pressure calculated by GOTHIC is 60.9 psia (which corresponds to the design pressure, 46 psig); CONTEMPT calculated a value of 60.5 psia. GOTHIC predicts nearly the same vapor temperature and fan cooler heat removal as CONTEMPT.

The sump liquid temperature calculated by GOTHIC is higher than the sump temperature calculated by CONTEMPT LT/28. This is due, as in the LOCA case discussed in Section 3.2.3 of this SER, to the more mechanistic method of modeling break flow drops and interfacial heat and mass transfer in GOTHIC.

WPSRSEM-NP, Revision 3 states that the calculated MSLB peak temperature results from a 1.1 ft^2 break in a main steamline while operating at 0 percent power. This analysis assumed the loss of one train of safeguards (two FCUs and one containment spray train). GOTHIC predicts nearly the same pressure. However, it underpredicts the vapor temperature between 50 seconds and the start of containment spray at 130 seconds in comparison with CONTEMPT. This is due to the drop size input value assumed in GOTHIC. Since the drop size used is conservative with respect to data (see Section 3.2.8 of this safety evaluation), the NRC staff finds this difference to be acceptable.

On a best estimate basis, the GOTHIC qualification report has shown GOTHIC to compare well with data and analytical solutions. In addition, the comparison for the MSLB of the GOTHIC evaluation model for Kewaunee with the CONTEMPT code, described in WCAP 15427 Revision 1, demonstrates an acceptable level of agreement. Therefore, the NRC staff finds the use of the Kewaunee GOTHIC evaluation model to be acceptable for calculating the containment response to MSLBs.

7.0 CONCLUSION FOR GOTHIC

Based on the evaluation given above, the NRC staff finds the licensee's use of the proposed GOTHIC evaluation model presented in WCAP 15427 Revision 1 and the licensee's April 13, and July 26, 2001, letters, to be acceptable for calculating the containment peak pressure and temperature response to a design-basis LOCA and design-basis MSLB.

The licensee did not present any description of how the GOTHIC code would be applied in performing subcompartment analyses. Therefore, the evaluation model presented in WCAP 15427 Revision 1 and the April 13, and July 26, 2001, letters, is not approved for subcompartment calculations.

The licensee has stated that the Kewaunee GOTHIC containment evaluation model will not be used to calculate the minimum pressure for LOCA backpressure analyses required for demonstrating compliance with the criteria of 10 CFR 50.46. Therefore, the Kewaunee evaluation model presented in WCAP 15427 Revision 1 and the April 13, and July 26, 2001, letters, is not approved for subcompartment calculations.

The NRC staff also finds the licensee's mass and energy release entrainment model used for the MSLB calculations to be acceptable for the reasons given in Section 3.5.2 of this SER. This resolves an unreviewed safety question raised in a NRC staff letter to the licensee dated April 15, 1998.

Based on all of the above, we find the reload topical report WPSRSEM-NP, Rev. 3, to be acceptable for containment peak pressure and temperature calculations for the design-basis LOCA and the design-basis MSLB.

8.0 REFERENCES FOR RETRAN

1. Letter from K. H. Weinhauser (NMC) to U. S. Nuclear Regulatory Commission, "Wisconsin Public Service Corporation Reload Safety Evaluation Methods Topical Report, WPSRSEM-NP, Revision 3," dated October 12, 2000.

2. Letter from M. E. Reddemann (NMC) to Document Control Desk (NRC), "Nuclear Management Company, LLC. Response to NRC's Request for Additional Information on Wisconsin Public Service Corporation Reload Safety Evaluation Methods Topical Report, WPSRSEM-NP, Revision 3," dated February 7, 2001.

3. Letter from M. E. Reddemann (NMC) to Document Control Desk (NRC), "Nuclear Management Company, LLC. Revised Response to NRC's Request for Additional Information on Wisconsin Public Service Corporation Reload Safety Evaluation Methods Topical Report, WPSRSEM-NP, Revision 3," dated March 7, 2001.

4. Letter from C. R. Steinhardt (WPSC) to Document Control Desk (NRC) transmitting WPSRSEM-NP-A, Revision 2, "Reload Safety Evaluation Methods for Application to Kewaunee," dated November 9, 1988.

5. Letter from J. G. Giiter (NRC) to D. C. Hintz (WPSC), “Wisconsin Public Service Corporation “Safety Evaluation Methods for Application to Kewaunee”,” dated April 11, 1988.

6. Letter from S. A. Richard (NRC) to G. L. Vine (EPRI), “Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow System" (TAC NO. MA4311),” dated January 25, 2001.

7. Letter from R. L. Laufer (NRC) to M. L. Marchi (WPSC), “Safety Evaluation for the High Thermal Performance (HTP) Departure from Nucleate Boiling (DNB) Correlation and the Associated 1.14 Minimum DNBR Safety Limit for Kewaunee Fuel,” dated December 30, 1997.

8. WCAP-10054-P-A, “Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code,” dated August 1985.

9. WCAP-10924-P-A, Revision 1, “Westinghouse Large-Break LOCA Best Estimate Methodology,” dated December 1988.

10. Letter from R. L. Laufer (NRC) to M. L. Marchi (WPSC), “Exemption from Certain Requirements of 10 CFR Part 50, Appendix K, Paragraphs I.D.3 and I.D.5 - Kewaunee Nuclear Power Plant (TAC No. M96132),” dated November 19, 1996.

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