NRC-03-108 10 CFR 50.90

November 5, 2003

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

KEWAUNEE NUCLEAR POWER PLANT
DOCKET 50-305
LICENSE No. DPR-43
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION AND SUPPLEMENTAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST 195, STRETCH POWER UPRATE FOR KEWAUNEE NUCLEAR POWER PLANT

References:
1) Letter NRC-03-057 from Thomas Coutu to Document Control Desk, "License Amendment Request 195, Application for Stretch Power Uprate for Kewaunee Nuclear Power Plant," dated May 22, 2003.

2) Letter to Mr. Thomas Coutu from John G. Lamb, "Kewaunee Nuclear Power Plant – Request for Additional Information for Proposed Amendment Request Regarding the Application for Stretch Power Uprate (TAC NO. MB9031)," dated October 7, 2003.

3) NRC Letter, "Kewaunee Nuclear Power Plant – Issuance of Amendment (TAC NO. MB6408)," dated September 29, 2003, approving License Amendment 169 authorizing use of GOTHIC 7 for Containment Design Basis Accident Analysis.

In accordance with the requirements of 10 CFR 50.90, Nuclear Management Company, LLC (NMC) submitted license amendment request (LAR) 195 (Reference 1) for a stretch power uprate of six percent. The stretch power uprate would change the operating license and the associated plant Technical Specifications (TS) for the Kewaunee Nuclear Power Plant (KNPP) to reflect an increase in the rated power from 1673 MWt to 1772 MWt.

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On October 7, 2003, the Nuclear Regulatory Commission (NRC) issued requests for additional information (RAIs) regarding the proposed stretch power uprate (Reference 2). This letter, with attachments and enclosures, contains the NMC responses to the NRC formal RAIs. In addition, this letter makes an editorial change to our Technical Specification bases submitted as part of Reference 1, adds a supplement to our Containment Integrity Analysis, adds a supplement to our Loss of Normal Feedwater (LONF) and Loss of AC Power (LOAC) Safety Analyses, and adds one new regulatory commitment to our original submittal. The following table summarizes the attachments to this letter.

| Attachment | Content Description |
|---|---|
| 1 | Responses to the requests for additional information, plus applicable enclosures (A through I). |
| 2 | TS bases pages marked up to show the additional proposed changes. |
| 3 | Revised (clean copy) TS bases pages. |
| 4 | Supplement to Containment Integrity Safety Analysis. |
| 5 | Supplement to Loss of Normal Feedwater (LONF) and Loss of AC Power (LOAC) Safety Analyses. |
| 6 | Revised List of Regulatory Commitments |

As Enclosure D contains information proprietary to Westinghouse Electric Company, it is supported by an affidavit signed by Westinghouse (Enclosure F), the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.790 of the Commission's regulations. Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 10 CFR 2.790. Correspondence with respect to the copyright or proprietary aspects of the item listed above or supporting the Westinghouse Affidavit, should reference the appropriate authorization letter and be addressed to H.A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

The changes to the Technical Specifications (TS) bases (Attachments 2 and 3) are editorial changes only to the proposed bases for the stretch power uprate. One change corrects a TS bases reference to a re-numbered TS and the other enhances the wording for a revised bases regarding containment cooling.

Attachment 4 is a supplement to our Containment Integrity Analysis for the Stretch Power Uprate. The analysis was re-performed using GOTHIC 7, but with NRC requested restrictions (Reference 3). This supplement documents that the containment responses to analyzed accidents remain acceptable at stretch uprate conditions.

Attachment 5 is a supplement to our Loss of Normal Feedwater (LONF) and Loss of AC Power (LOAC) analyses for the Stretch Power Uprate. The analyses were re-performed after it was discovered that a least negative Doppler-only power coefficient (DPC) expression was inadvertently assumed, rather than the most negative DPC expression. These safety analyses supercede previously reported safety analyses for the stretch power uprate, and indicate that all applicable acceptance criteria are met.

Attachment 6 is a revised copy of regulatory commitments listed in our submittal (Reference 1) as Attachment 7. We are adding a new commitment as a result of our response to RAI question #57. The new commitment will be #16. None of the original commitments are changed.

These responses to the RAIs, revised Technical Specification Bases, and supplements do not change the Operating License or Technical Specifications for the KNPP, nor do they change any of the proposed changes to the Operating License or Technical Specifications in reference 1. Attachments 2 and 3 only make a change to the Technical Specification Bases. This response also does not change the no significant hazards determination, the environmental considerations, the requested approval date, or the requested implementation period originally submitted in reference 1.

In accordance with 10 CFR 50.91, a copy of this letter, with attachments, is being provided to the designated Wisconsin Official.

If there are any questions or concerns associated with this response contact Mr. Gerald Riste at (920)388-8424

I declare under penalty of perjury that the foregoing is true and correct.
Executed on November 5, 2003.

Thomas Coutu

Thomas Coutu
Site Vice-President, Kewaunee Plant

BJW

Attachments:
1. Responses to Requests for Additional Information
2. TS bases pages TS B3.3-3 and TS B4.8-1 marked up to show the proposed changes
3. Revised (clean copy) TS bases pages TS B3.3-3 and TS B4.8-1
4. Supplement to Containment Integrity Safety Analysis
5. Supplement to Loss of Normal Feedwater (LONF) and Loss of AC Power (LOAC) Safety Analyses
6. Revised List of Regulatory Commitments

Enclosures:

A. American Transmission Company, Facility Study Report (Interim), Generator Interconnection Request GIC050 (G165) (MISO # 37239-01), 38 MW Increase at the Kewaunee Nuclear Generation Facility Kewaunee County, Wisconsin, October 13, 2003
B. WCAP 8339, Westinghouse Emergency Core Cooling System Evaluation Model – Summary, June, 1974
C. Westinghouse Technical Bulletin, NSID-TB-86-08, Post-LOCA Long-Term Cooling: Boron Requirements, October 31, 1986
D. WCAP-15821-P, Westinghouse Protection System Setpoint Methodology Kewaunee Nuclear Plant (Power Uprate to 1757 MWt-NSSS Power with Feedwater Venturis, or 1780 MWt-NSSS Power with Ultrasonic Flow Measurements, and 54F Replacement Steam Generators), May 2003, Proprietary Version
E. WCAP-15821-NP, Westinghouse Protection System Setpoint Methodology Kewaunee Nuclear Plant (Power Uprate to 1757 MWt-NSSS Power with Feedwater Venturis, or 1780 MWt-NSSS Power with Ultrasonic Flow Measurements, and 54F Replacement Steam Generators), October 2003, Non-Proprietary Version
F. Westinghouse authorization letter, CAW-03-1726, an accompanying affidavit, proprietary information notice, and copyright notice for Enclosure D
G. Kewaunee Procedure GNP-04.06.01, Revision E (December 19, 2002), Plant Setpoint Accuracy Calculation Procedure
H. Excerpt, Generic Section of Kewaunee I&C Calculations, Methodology
I. KNPP Concern No. 92006-02 (4/9/92), documenting a question of the fault ratings on 4.16KV buses, and the subsequent resolution

cc- US NRC, Region III
US NRC Senior Resident Inspector (w/o enclosures)
Electric Division, PSCW ( w/o enclosures)

ENCLOSURE F

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

**November 5, 2003**

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Responses to Requests for Additional Information and Supplemental Information Regarding LAR 195

Westinghouse authorization letter, CAW-03-1726, an accompanying affidavit, proprietary information notice, and copyright notice for Enclosure D

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Direct tel: (412) 374-5036
Direct fax: (412) 374-4011
e-mail: Galem1js@westinghouse.com

Our ref: CAW-03-1726

October 30, 2003

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-15821-P, "Westinghouse Protection System Setpoint Methodology Kewaunee Nuclear Plant (Power Uprate to 1757 MWt-NSSS Power with Feedwater Venturis, or 1780 MWt-NSSS Power with Ultrasonic Flow Measurements, and 54F Replacement Steam Generators)" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-03-1726 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Nuclear Management Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-03-1726 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: J. Dyer
D. Holland
B. Benney
E. Peyton

bcc: H. A. Sepp (ECE 4-7A) 1L
R. Bastien, 1L, 1A (Nivelles, Belgium)
C. Brinkman, 1L, 1A (Westinghouse Electric Co., 12300 Twinbrook Parkway, Suite 330, Rockville, MD 20852)
RCPL Administrative Aide (ECE 4-7A) 1L, 1A (letter and affidavit only)
R. Owoc (ECE 419J) 1L, 1A
H. Hanneman (NMC) 1L, 1A
J. Holly (NMC) 1L, 1A
G. Riste (NMC) 1L, 1A

CAW-03-1726

# AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared H. A. Sepp, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

H. A. Sepp

H. A. Sepp, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 30th day
of October, 2003

Sharon L. Fiori

Notary Public

SHARON L. FIORI
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company LLC.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Electric Company LLC in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-15821-P, "Westinghouse Protection System Setpoint Methodology Kewaunee Nuclear Plant (Power Uprate to 1757 MWt-NSSS Power with Feedwater Venturis, or 1780 MWt-NSSS Power with Ultrasonic Flow Measurements, and 54F Replacement Steam Generators)" (Proprietary), dated May 2003 for Kewaunee Nuclear Plant, being transmitted by the Nuclear Management Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse Electric Company LLC for Kewaunee Nuclear Plant is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification of plant power uprating.

This information is part of that which will enable Westinghouse to:

(a) Provide information in support of plant power uprate submittals.

(b) Provide plant specific calculations.

(c) Provide licensing documentation support for customer submittals.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation associated with power uprate submittals.

(b) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculations, evaluations, analysis, and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

## PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

ATTACHMENT 1

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

**November 5, 2003**

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Responses to Requests for Additional Information and Supplemental Information Regarding LAR 195

Responses to Requests for Additional Information

62 Pages to Follow

**Questions from the Materials and Chemical Engineering Branch (EMCB) – Piping Integrity and NDE Section**

**Increased power increases the potential for materials degradation of Nuclear Steam Supply System (NSSS)/ Balance of Plant (BOP) piping. With respect to this issue provide the following information:**

1. **Please discuss the determination made for service adequacy of the materials in the NSSS/BOP piping with increased temperature and pressure due to the power uprate.**

   **NMC Response:**

   Service adequacies of piping materials for the NSSS piping systems are discussed in Attachment 4 to the application, Sections 5.5, Reactor Coolant Loop Piping and Supports, and 5.9, Nuclear Steam Supply System Auxiliary Equipment. Service adequacies of the materials in the BOP piping are discussed in our responses to questions #54 through #56.

2. **Please discuss the determination made for service adequacy of the materials in the control rod drive mechanisms taking into consideration Bulletins 2002-01 and 2002-02.**

   **NMC Response:**

   NRC Bulletins 2002-01 (Reactor Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity) and 2002-02 (Reactor Vessel Head and Vessel Head Penetration Nozzle Inspection Programs) address reactor coolant pressure boundary integrity and pressure vessel head and nozzle degradation. Both bulletins identify the regulatory basis for reactor coolant pressure boundary integrity that must be satisfied as part of the design basis.

   NRC Bulletin 2002-01 requires holders of operating licenses for pressurized water reactors to evaluate the ability of the existing inspection and maintenance programs to identify degradation of the reactor pressure vessel head and to identify any degradation observed and corrective actions taken, including root cause determination. NRC Bulletin 2002-02 requires holders of operating licenses for pressurized water reactors to either provide a summary of supplemental inspections to be implemented to detect reactor vessel head and nozzle degradation, or to justify continued reliance on visual inspections to detect degradation.

   Attachment 4 of our submittal, Sections 5.1 and 5.4 provide the structural evaluations of the reactor vessel (including the reactor vessel head) and the Control Rod Drive Mechanisms (CRDMs), respectively, for the power uprate operating conditions.

   The structural evaluations performed in support of the Power Uprate Project for these components confirmed that the impact of the Performance Capability Working Group (PCWG) design operational parameters, and the Nuclear Steam Supply System (NSSS) design transients for the Power Uprate Program, are bounded by the parameters and transients considered for either the Replacement Steam Generator Program or the

original licensing design analysis. The intent of the structural evaluations for the Power Uprate Program is to ensure that the reactor pressure boundary components for the reactor vessel, reactor vessel head, and CRDMs continue to satisfy the current licensing basis for reactor coolant pressure boundary integrity, including the requirements specified in these two NRC Bulletins, and the ASME Boiler and Pressure Vessel Code of record, following implementation of the Power Uprate Project.

The Power Uprate Project does not include any material changes to either the reactor vessel, reactor vessel head, or CRDMs. As noted above, the revised operating parameters for power uprate have been evaluated and shown to be acceptable. Therefore, the licensing basis over the remaining service lifetime of these components continues to remain valid.

NMC is planning to replace the Kewaunee reactor vessel head, tentatively scheduled for the fall 2004 refueling outage. The replacement reactor vessel head will be installed with penetrations made of inconnel alloy 690 tubing and alloy 52 weld metal. Following the replacement, NMC will assess the need to perform future reactor vessel head examinations.

If, due to unforeseen schedule or technical difficulties, the reactor vessel head is not replaced during the fall 2004 outage, the existing head will be in the high susceptibility category of NRC Order EA-03-09 (>12 EDY). NMC will perform all required non-destructive exams outlined in NRC order EA-03-09 prior to placing the reactor vessel head back in service.

## Questions from the EMCB Branch – SG Integrity and Chemical Engineering Section

3. **In Section 5.7.10, "Tube Repair Limits (Regulatory Guide 1.121 Analysis)" of Attachment 4 to the application, the licensee indicates that an analysis has been performed to define the structural limits for an assumed uniform thinning in both the axial and circumferential directions. Calculations have also been performed to establish the structural limit for tube straight leg (free span) flaws over an unlimited axial extent and for degradation over limited axial extent at the tube support plate and anti-vibration bar intersections. The licensee does not conclude whether the SG tube repair limits in the technical specifications are still acceptable given the results of the calculation discussed above. Therefore, the staff requests the licensee state and explain whether the current repair limits in the technical specifications remain appropriate for operation under the 6 percent uprated power conditions.**

**NMC Response:**

Tube structural limits are defined for the Kewaunee replacement steam generators in WCAP-15325, "Regulatory Guide 1.121 Analysis for the Kewaunee Replacement Steam Generators." A revised analysis was performed to document applicable tube structural limits for the stretch power uprate (SUR) conditions. The analysis results show that, although the primary-to-secondary pressure gradients are increased for the SUR conditions, the changes were not large enough to result in a change to the structural limits. As a result, the tube structural limits, as defined in Table 6-1 of WCAP-15325, remain applicable to for KNPP at SUR conditions.

The KNPP tube repair limits are calculated from the tube structural limits by subtracting an allowance for eddy current uncertainty and continued growth. Eddy current uncertainty and allowances for growth are a function of the NDE technique employed and are not impacted by operation at SUR conditions.

As the tube structural limits remain applicable for KNPP at SUR conditions, and as the eddy current uncertainty and allowances for growth are not impacted by operation at SUR conditions, it is concluded that the current repair limits in the KNPP technical specifications remain appropriate for operation under the 6% uprated power conditions.

## Questions from the Plant Systems Branch (SPLB) – Fire Protection Section

4. **Although the license amendment request includes a discussion of impact on the ability to reach cold shutdown, the license amendment request does not include any discussion regarding changes to the fire protection (FP) program or other operating conditions that may adversely impact the post-fire safe shutdown capability in accordance with Appendix R. Clarify whether this license amendment request involves changes to the FP program or other operating conditions that may adversely impact the post-fire safe shutdown capability in accordance with Appendix R. Provide the technical justification for changes.**

   NMC Response:

   The KNPP Fire Protection (FP) program, the FP program design description, and the FP program bases have been reviewed for Stretch Uprate (SUR) operating conditions. Based on this review, the license amendment request for SUR (KNPP Letter #NRC-03-057 for LAR 195, 5/22/03 ) does not involve physical changes to the FP program, and does not impact the post-fire safe shutdown capability in accordance with Appendix R. Because of increased reactor decay heat and equilibrium Xenon concentration for the SUR operating conditions the following FP program design basis analyses are being evaluated and/or analyzed, and the Appendix R and Fire Protection system design description and associated procedures updated appropriately prior to implementation of the SUR:

   1) Appendix R cool down was analyzed for SUR operating conditions. Reference Attachment 4, section 4.1.4.3.2 of our submittal. The required appendix R cool down time of 72 hours is demonstrated to be satisfied at SUR operating conditions. KNPP Appendix R procedures were revised to ensure adequate component cooling water system flow consistent with the assumptions of the Appendix R cool down analysis.

   2) The time to SG dry out following a loss of feedwater is reduced slightly due to higher decay heat at SUR operating conditions. The time to SG dry out will be re-evaluated for SUR operating conditions. The re-evaluation is to update program documents for the SUR, and to verify current operator response times for auxiliary feedwater system and charging system initiation continue to be less than the time to SG dry out. This evaluation, and update of any required documents, will be completed prior to implementation of the SUR. This is

considered part of our Regulatory Commitment #12 (Reference Attachment 7 of our submittal).

3) Requirements associated with borating the RCS to cold shutdown, utilizing the RWST, with and without letdown in service are being evaluated for any required procedure changes. This includes verifying available pressurizer volume to accommodate the boration in the pressurizer if letdown is not available. This evaluation, and update of any required procedures, will be completed prior to implementation of the SUR. This is considered part of our Regulatory Commitment #12 (Reference Attachment 7 of our submittal).

## Questions from the Electrical & Instrumentation and Controls Branch (EEIB) – Electrical Engineering Section Section

5. **Provide details about the grid stability analysis including assumptions and results and conclusions for the stretch power uprated condition.**

**NMC Response:**

Enclosure A contains the grid stability analysis performed by American Transmission Company, LLC (ATC) to evaluate the impact of the proposed KNPP power uprate. This study, "Facility Study Report (*Interim)* Generator Interconnection Request GIC050 (G165) (MISO #37239-01) 38 MW Increase at the Kewaunee Nuclear Generation Facility Kewaunee County, Wisconsin," dated October 13, 2003, evaluated a 38 megawatt electric (MWe) uprate implemented in two phases: a 10 MWe addition in May 2003 and the remaining 28 MWe addition in March 2004. The first phase has already been completed with the implementation of the 1.4% MUR power uprate in July 2003. The second phase is currently planned for implementation in March 2004, after receipt of NRC approval of License Amendment Request 195 for the 6% stretch power uprate.

This Facility Study determined that several transmission system upgades are required to address pre-existing stability issues, which will be corrected prior to implementation of the second phase (28 MWe) of the KNPP power uprate. These transmission system upgrades (e.g., replacement of four breakers at the North Appleton substation and modification of three relays at the Rocky Run substation) or interim solutions (special protection schemes) will be installed by ATC prior to implementation of the 6% stretch power uprate. Furthermore, existing ATC transmission system operating guides with restrictions on KNPP generation with transmission lines out of service may also be revised based on additional studies scheduled to be completed by ATC in 2003. The required installation of transmission system upgrades and possible revision of transmission system operating guides has been captured in the KNPP site corrective action process to be completed prior to implementation of the stretch power uprate, and will be documented as a prerequisite in the uprate implementation plan.

6. **Provide in detail the effects of the stretch power uprate on the station blackout coping capability. The evaluation should address the capacities of the condensate storage tank, turbine driven auxiliary feedwater pump, station batteries, and backup air supplies for air operated valves for decay heat removal and RCS cooldown during the time period of an SBO.**

**NMC Response:**

Station Blackout (SBO) coping and mitigation requirements are specified in the USNRC SBO rule, 10CFR 50.63. Guidelines for SBO coping are provided in NUMARC 87-00, Revision 1, except where Regulatory Guide 1.155 is more restrictive. In accordance with these documents, KNPP is subject to a four hour SBO coping duration.

Decay heat removal is accomplished by powering the Turbine Driven Auxiliary Feedwater Pump from either Steam Generator (SG), discharging Condensate Storage Tank (CST) water to the 1A and 1B SGs, and discharging steam from the 1A and 1B SG Power Operated Relief Valves (PORVs). The current KNPP Technical Specifications require a minimum useable CST inventory of 39,000 gallons during power operation. Power uprate results in increased decay heat, so the minimum useable CST inventory is impacted. A new SBO event analysis was performed as part of the balance of plant (BOP) evaluations to determine the required CST inventory for the power uprate. The results of the power uprate analysis show that the required CST minimum inventory must be raised from 39,000 gallons to 41,500 gallons. The flow path and valve requirements , including station battery power and nitrogen backup supply for air operated valves (AOV), for accomplishing decay heat removal and RCS inventory control are not impacted by power uprate and remain the same as described in the original SBO mitigation strategy.

Reactor Coolant System (RCS) inventory control is accomplished by the use of a single Charging Pump drawing suction from the Refueling Water Storage Tank (RWST) and discharging to the Reactor Coolant Pump seals and RCS loop B cold leg. The flow path and valve requirements, including station battery power and nitrogen backup for air operated valves (AOV), for accomplishing RCS inventory control are not impacted by power uprate and remain the same as described in the original SBO mitigation strategy.

KNPP has two emergency diesel generators. If all normal power sources to the emergency buses should fail, the two diesel generators, one connected to ESF 4160-V bus 1-5 and one connected to ESF 4160-V bus 1-6, provide emergency power. In the event of a Station Blackout, either one of the emergency diesel generators has the capability to withstand and recover from the SBO within the four-hour coping duration in accordance with the requirements of Regulatory Guide 1.155. The capability to provide sufficient emergency power to buses 1-5 or 1-6 in a SBO within the four- hour coping period is not impacted by power uprate.

An Alternate AC (ACC) power source will utilize the Technical Support Center (TSC) diesel generator. This diesel generator is an independent, non-class 1E 600 kW diesel generator that provides emergency power to 480-V Bus 1-46 for TSC equipment. For SBO purposes, a connection can be made between this bus and the 480-V safety ESF Bus 1-52. A class 1E breaker at Bus 1-52 and a non-class 1E breaker at Bus 1-46 provide isolation between the two buses during normal operation. For SBO, selected

non-essential loads will be stripped from each of the two buses, and the two breakers will be closed to provide power to essential loads on both buses. The alternate AC power source is determined to be adequate to provide sufficient power to essential loads on buses 1-52 and 1-46 to mitigate SBO at power uprate conditions.

Evaluations of the systems impacted by the uprate did not identify any changes to design or operating conditions that adversely affect the ability to provide safe shutdown for a SBO initiated from stretch power uprate (SUR) conditions. The total volume of water that would be supplied by the Auxiliary Feedwater (AFW) System will be increased slightly due to power uprate. However, there is sufficient useable capacity in the Condensate Storage Tank (or Service Water, which is available to replenish AFW inventory) to meet hot standby and cooldown requirements required by the Technical Specifications and those imposed by the plant design basis.

The KNPP Station Blackout Mitigation Design Description and the ability of KNPP to cope with a SBO event have been evaluated for the impact of the stretch power uprate to 1772 MWt reactor power. Based on this evaluation, the capabilities of existing plant systems, structures, and components (SSCs) are adequate for removal of decay heat and inventory control of the RCS with the plant at the SUR conditions. The only plant change that was required as a result of SBO evaluation and/or re-analysis was an increase in the CST Technical Specification inventory from 39,000 gallons to 41,500 gallons (Reference Attachment 1 to the application, Section 3.5, and Attachment 4, Section 8.3.9). With this one change, all SBO requirements are satisfied at the power uprate conditions.

7. **With the stretch power uprate, the megavolt-amperes reactive (MVARS) supplied by the main generator (MG) is reduced which affects the voltages at the plant. Explain the effects on the voltage at the plant and how this affects plant equipment. Explain how you will ensure proper voltages.**

**NMC Response:**

As stated in WCAP-16040-P, Section 8.3.14.1.1.3, the generator output at power uprate is within the existing generator capability curve. The existing generator, exciter and cooling equipment are adequate to support unit operation at power uprate conditions.

Once the generator is connected to the grid, the nominal output voltage of the generator to the electrical grid is determined by the grid, and can only be varied by our generator a very small amount. Therefore, the voltages at the plant remain unchanged. While connected to the grid, KNPP can vary the power (MVARs) supplied to the grid. The grid voltage and power requirements are maintained by Energy Supply and Control, and the grid voltage range maintained will not change after the stretch power uprate.

Since operation will remain within the existing capability curve, and voltages at the plant will not change, there are no adverse effects on voltage or operation of plant equipment. If Energy Supply and Control requests an increase in MVARs to support grid voltage, a power reduction would be required to stay within the generator capability curve.

**8. The licensee stated the following:**

> **"The Main Transformer (MT) is not capable of supporting station operation at full power uprate conditions with the main generator operating in the leading mode with the hotel load supplied by the Reserve Auxiliary Transformer (RAT). Under this operational scenario, the maximum amount of Reactive Power that can be accepted, measured at the MT secondary, is limited to 262 MVARS. Under these power uprate conditions, the MT operates within its 65°C rating."**

**Explain the operation of the MT describing what loads it is supplying during startup, shutdown, and normal operation. Explain the operation of the RAT describing what loads it is supplying during startup, shutdown, and normal operation. Explain the interrelation between the MT and the RAT. Explain in detail the effects on operation and equipment if you operate in the leading mode greater than 262 MVARS. Explain in detail how you will control plant operation so you will never exceed the limit of 262 MVARS in the leading mode. Explain in detail any other limitations.**

**NMC Response:**

The generator 3-phase output is stepped up from 20 kV to 345 kV by the three Main Transformers, one per phase. The 345 kV power is sent to the switchyard from the high side of the Main transformers via the generator output breaker G-1. The generator output is also supplied to the Main Auxiliary Transformer (MAT) to step the 20 kV down to 4160 VAC. Split windings on the MAT secondary can supply all six in-plant 4160 VAC buses. The MAT normally supplies 4160 VAC power to in-plant buses 1, 2, 3, and 4 when greater than 15% power.

The Reserve Auxiliary Transformer (RAT) receives 138 kV from the switchyard and normally provides 4160 VAC to Safeguards Bus 6. At less than 15% power, and during plant shutdown, the RAT also provides power to Buses 1 through 4.

The Tertiary Auxiliary Transformer (TAT) receives 13.8 kV from the switchyard and normally provides 4160 VAC to Safeguards Bus 5 during normal and shutdown conditions.

The Main Transformer provides power to the switchyard from generator output. The RAT provides power to in-plant loads from the switchyard. The interrelationship, and basis for the above restriction, is whether or not the MAT is supplying house loads and reducing the loading on the MT.

Plant procedures restrict operation to within the generator capability curve. At the predicted full power load of 595.7 MWe, the maximum allowable reactive power, per the generator capability curve, is approximately 180 MVARs in the leading mode. Therefore, maintaining operation within the generator capability curve prevents exceeding the limit of 262 MVARs at the Main Transformer. The 262 MVAR limit on the MT was from a bounding analysis on the transformer that ended up outside the generator capability curve.

**9. The licensee stated the following:**

> **"The MT has limited capability to support power uprate with the main generator operating in leading mode and the hotel load supplied from the RAT. Uprate generator operation in the underexcited region, when the hotel load is supplied from the Tertiary Auxiliary Transfomer (TAT) and the RAT, reactive load will be limited to 250 MVAR or less in order to avoid overloading the MT at the 65°C rating."**

**Explain the operation of the TAT describing what loads it is supplying during startup, shutdown, and normal operation. Explain the interrelation between the MT, the RAT, and TAT. Explain in detail the effects on operation and equipment if you operate in the leading mode greater than 250 MVARS. Explain in detail how you will control plant operation so you will never exceed the limit of 250 MVARS in the leading mode. Explain in detail any other limitations.**

**NMC Response:**

See previous response to question #8 for operation of the MT, RAT, and TAT.

As stated in the response to question #8, maintaining operation within the generator capability curve prevents exceeding the limit of 250 MVARs at the Main Transformer.

**10. The licensee stated that the reactor coolant pumps (RCP) and feedwater (FW) pumps voltages are within the operational limits of NEMA MG-1. Explain the effects of operating the RCPs and FW pump motors within the operational limits of NEMA MG-1. Justify the acceptable operation of the RCPs and FW pump motors within the operational limits of NEMA MG-1. Describe your intended operation of the RCPs and FW pump motors with respect to the stretch power uprate. If you intend to operate the RCPs and FW pump motors with reduced voltage grid conditions under the stretch power uprate conditions, please justify the acceptable operation.**

**NMC Response:**

Section MG 1-20.45 VARIATIONS FROM RATED VOLTAGE AND RATED FREQUENCY, states that induction machines (motors) shall operate successfully under running conditions at rated load with a variation in the voltage up to plus or minus ten percent of rated voltage (20.45.1 Running). The rated voltage of these motors is 4000 Volts. Therefore, the acceptable voltage range is 3600 to 4400 Volts. Operation at 3965 Volts, as predicted, is within 1% of the ideal operating point of 4000 Volts.

The stretch power uprate is calculated to cause a voltage reduction of 12 Volts on the 4160 Volt (nominal) buses feeding the RCPs and FW pump motors due to the additional horsepower load on the FW pump motors. This is a 12V/4160V = 0.0029%, or about 0.3% change and is not significant.

The RCPs and FW pump motors are not intended to be operated at reduced grid voltage conditions due to the stretch power uprate.

For buses 1-1 and 1-2 and FW pump motors, see page 8-74 of WCAP-16040-P, and answer to previous question NRC #10 above on FW pump motors.

There is no stretch uprate load flow analysis for buses 1-1 and 1-2. The current load flow analysis and supplemental calculations, as described in section 8.3.14.1.6 of WCAP-16040-P, show acceptable results.

13. **The licensee stated the following:**

> **"Based on the equipment ratings and the calculated fault currents for the current plant condition, the fault current available, non-safety medium voltage buses 1-1 through 1-4 and their associated circuit breakers are overdutied. This condition was previously evaluated and found acceptable."**

**Explain the effects on buses 1-1 through 1-4 and their associated circuit breakers if they are overdutied. Explain in detail the justification for this overdutied condition and the acceptance criteria used. Please provide the evaluation that found the overdutied condition acceptable.**

**NMC Response:**

If a maximum possible fault current situation actually occurred, the breaker attempting to interrupt the fault could fail catastrophically.

The justification is that actual test data taken for the circuit breakers demonstrated that they could interrupt fault current greater than the maximum available fault current at KNPP.

The requested evaluation is provided as Enclosure I.

14. **The licensee stated the following:**

> **"The station has a spare RAT. The spare, when installed, provides increased fault current. With the spare RAT installed, buses 1-1 and 1-2 are enveloped by the normal case. Buses 1-3 and 1-4 are not enveloped by the normal case. Buses 1-5 and 1-6 are overdutied with the spare RAT feeding the buses. This condition of increased fault current availability was addressed in the fault current analysis that indicates the spare RAT should not be installed "without some provision for reducing fault current on its low-voltage side"."**

**Do you plan to operate with the spare RAT feeding the buses under the stretch power uprate? If so, justify the acceptable operation of buses 1-5 and 1-6 in an overdutied condition and explain the provisions for reducing fault current on the spare RAT's low-voltage side. Explain your intended operation under the stretch power uprate.**

**NMC Response:**

No, the spare RAT will only be installed on a failure of the installed RAT that cannot be repaired in place.

If the spare ever had to be installed, provisions for reducing the fault current to protect the safeguards buses might include the standard practice of a current limiting reactor (choke) installation, or other approved method. This would be decided at the time the spare RAT might be required. This decision would be part of the engineering evaluation required by plant procedures to install the spare RAT. Since the spare RAT is not identical to the installed RAT, it would not be a like-for-like replacement, and would require an engineering evaluation prior to use. The currently installed RAT would then replace the spare at the next feasible opportunity following repair.

**15. The licensee stated the following:**

> **"The increased load resulting from the increased motor loads on buses 1-1 through 1-4 will reduce bus voltage from its current operating value at power uprate. This reduction in the actual pre-fault voltage will result in a decrease in fault current available at the power uprate condition with respect to the current operating point."**

**Explain the relationship between the startup current and voltages at the stretch power uprate conditions. Explain the relationship between the fault currents and fault voltages at the stretch power uprate conditions. Justify the acceptable operation at the stretch power uprate condition with respect to the reduction in the actual pre-fault voltage and the decrease in the fault current available.**

**NMC Response:**

Startup current and voltage will not change for the stretch power uprate. The additional load will not be present on any large motor during plant startup.

The lower prefault voltage at stretch power uprate conditions will result in a lower calculated fault current.

The lower fault current and voltage will decrease the potential for problems interrupting a fault on buses 1-1 through 1-4.

**16. The licensee stated that the RCPs, FW pumps and condensate pumps will operate above rating. Explain operation above rated horsepower and its impact in motor overheating and degradation of the windings. Provide justification for operating the RCPs, FW pumps and condensate pumps above rating. Also, explain how the relay operation is affected and justify its acceptable operation.**

**NMC Response:**

The reactor coolant pump motors are not predicted to operate above their rating after the stretch uprate. WCAP-16040-P (Attachment 4 to our submittal), Table 8.3.14.1-1, shows them operating at 5942 Hp versus their rating of 6000 Hp (as stated on pg 8-75).

The feedwater pump motors will operate at about 1.03 times full load Hp, well within their 1.15 service factor (see pgs 8-76 and 8-77 of WCAP-16040-P). NEMA MG-1 allowed rise for feedwater pump motors is 85°C (over 40°C ambient, Class B insulation), so allowed motor stator temperature is 125°C. The feedwater pump motors typically run at less than 110°C motor stator temperature.

The condensate pump motors will operate at about 1.02 times full load Hp, well within their 1.15 service factor (see pgs 8-77 and 8-78 of WCAP-16040-P). The uprate predicted motor temperature rise is 62.3°C versus NEMA MG-1 allowed rise of 90°C (over 40°C ambient), so allowed motor stator temperature is 130°C. The condensate pump motors typically run at less than 100°C motor stator temperature.

Overcurrent relay operation will not be challenged by the increased power (and current) drawn, as the relays are set at about 125% of motor full load amps (FLA). The increase in motor current should be less than 5% over FLA, leaving a 20% margin for voltage variation effects prior to the overcurrent relays going into alarm status.

**17. The licensee stated the following:**

> **"The EQ equipment inside containment will be evaluated to demonstrate the affected equipment is qualified for the EQ long-term temperature."**

**Provide evaluations that demonstrate the affected equipment is qualified for the EQ long-term temperature.**

**NMC Response:**

The evaluations that demonstrate the affected equipment inside containment is qualified for the EQ long term containment temperature profile are being performed. NMC will complete this action by 12/15/03, as requested by the NRC, to verify equipment qualification. This meets our commitment, as stated in our Stretch Power Uprate submittal, Commitment 9, of completing the appropriate evaluations prior to implementation of the uprate.

**18. The licensee stated the following:**

> **"For those components where the thermal lag temperatures exceeded the equipment qualification temperature, the EQ equipment required for HELB outside containment will be evaluated to demonstrate the affected equipment is qualified for the EQ thermal lag temperatures."**

**Provide evaluations that demonstrate the EQ equipment required for HELB outside containment is qualified for the EQ thermal lag temperatures.**

**NMC Response:**

The evaluations in question are being performed to demonstrate that for those components where the thermal lag temperatures exceeded the equipment qualification temperature, the EQ components required for HELB outside containment are qualified for the EQ thermal lag temperatures. NMC will complete these evaluations and verify

component qualification by 12/15/03, as requested by the NRC. This meets our commitment, as stated in our Stretch Power Uprate submittal, Commitment 11, of verifying the equipment is qualified prior to implementation of the uprate.

## Questions from Reactor Systems Branch (SRXB)

**REFERENCES:**

1. **Letter from T. Coutu, Nuclear Management Company, LLC (NMC), to USNRC, "License Amendment Request 195, Application For Stretch Power Uprate For Kewaunee Nuclear Power Plant," Docket 50-305, License No. DPR-43, dated May 22, 2003.**
2. **Letter from T. Coutu, Nuclear Management Company, LLC (NMC), to USNRC, "NMC Responses to NRC Request for Additional Information Concerning License Amendment Request No. 187 to the Kewaunee Nuclear Power Plant Technical Specifications (TAC No. MB5718)," Docket 50-305, License No. DPR-43, dated February 27, 2003.**

**19. The licensee's submittal (Reference 1) states that the FW control valves were modified to accommodate higher feedwater flow rates. Was this modification and the higher feedwater flow considered in the updated safety analysis report (USAR) Chapter 14 accident and transient analyses reviewed by the staff in the Reload Transition Safety Report (RTSR) for KNPP License Amendment Request (LAR) No. 187? Provide technical justification if not considered.**

NMC Response:

The feedwater regulating (control) valve (FWRV) modification and the resulting higher feedwater flow were considered in the USAR Chapter 14 accident and transient analyses reviewed by NRC in the Reload Transition Safety Report (RTSR) for KNPP LAR 187.

For most of the USAR Chapter 14 transient analyses reviewed by the NRC in the Reload Transition Safety Report (RTSR) for KNPP LAR 187, the feedwater regulating (control) valves (FWRVs) were not explicitly modeled, and the feedwater flow was assumed to match the steam flow. As the uprated power level requires an increased steam flow, there was a corresponding increase in feedwater flow that had been considered in the analyses. For the main steam line break (MSLB) mass and energy (M&E) release analysis used in the MSLB containment integrity analysis, a USAR Chapter 14 accident analysis was included in our submittal, in which the modified FWRVs (new valve characteristic $C_V$ curve) were explicitly modeled.

**20. Section 2.2 of the licensee's submittal letter (Reference 1) provides a general discussion of the Loss of Normal Feedwater Analysis performed to support the stretch power uprate. Based on this reanalysis, the licensee must implement new technical specification (TS) requirements for auxiliary feedwater (AFW) train operability. Because AFW is relied upon to mitigate other Chapter 14 accidents and transients, please discuss the impacts that the proposed AFW TS change will have on any other potentially effected USAR Chapter 14 events.**

**NMC Response:**

The USAR Chapter 14 accident and transient analyses directly impacted by the auxiliary feedwater (AFW) system operation are loss of normal feedwater (LONF), main steam line break (MSLB), and loss of coolant accident (LOCA). The LONF analysis at stretch uprate (SUR) conditions necessitated the AFW Technical Specification (TS) changes, and is therefore the bounding accident from the standpoint of AFW train operability at stretch uprate power.

All other USAR Chapter 14 accidents, directly impacted by the AFW system, are bounded by the greater AFW requirements of the LONF accident, and therefore, are not impacted by the AFW TS change. None of these other USAR Chapter 14 accidents and transients would have required an AFW TS change. It should be noted that the increased AFW requirements are <u>ONLY</u> for operation greater than 1673 MWt core power.

21. **The licensee's submittal letter (Reference 1) states that the full power $\Delta T_0$ inputs to the overtemperature delta T and the overpower delta T setpoints will be changed to the predicted values based on best estimate evaluations for the stretch uprated power (1772 MWt) condition.**

    a. **Please provide a more specific discussion regarding the actual changes being made. Include the exact changes being made.**

    **NMC Response:**

    The full power $\Delta T_0$ inputs to the OTDT and the OPDT setpoint functions are changed prior to increasing power for stretch uprate (SUR). The predicted reactor delta T ($\Delta T$) values are based on measured full power $\Delta T$ values for the current plant at 1673 MWt, increased linearly for the stretch uprate power (1772 MWt) operating conditions. $\Delta T_0$ is the indicated $\Delta T$ at rated power. The actual changes to be made are as follows:

    1. Increasing the nominal full power $\Delta T$ input signal from 58.5ºF to 62ºF to the OTDT and OPDT setpoint functions (actual $\Delta T$ input signal is specific for each coolant loop).

    2. In the OTDT setpoint, change the K1 constant and f($\Delta$I) function inputs constant with values from the Cycle 26 Reload Safety Evaluation (RSE) supplement for operation at 1772 MWt.

    b. **Provide a reference or discuss the methodology used to determine the predicted values.**

    **NMC Response:**

    The predicted SUR reactor delta $T_0$ ($\Delta T_0$) values are determined using measured full power $\Delta T_0$ values for the current plant at 1673 MWth and increasing them linearly (multiplying by 1772/1673) for the stretch uprate power (1772 MWt) operating conditions.

c. **Identify those USAR Chapter 14 transients which credit these trips and discuss how the proposed change impacts the analyses. Discuss the impacts on specific acceptance criteria for these events.**

**NMC Response:**

The transient analyses in which either the OTΔT or OPΔT reactor protection functions are credited, i.e., one of these trips actuates and mitigates the consequences of the accident or transient, are: Uncontrolled Rod Withdrawal at Power, Chemical and Volume Control System Malfunction, Loss of External Load, and Main Steam Line Break outside containment. Refer to Table 5.1-5 of Attachment B to our response (KNPP Letter #NRC-03-016 of February 27, 2003) to your Request For Additional Information (RAI) of January 21, 2003 (MB5718), and Table 6.5.4 of Attachment 4 to our application for Stretch Power Uprate.

The OTΔT and OPΔT setpoints assumed in the RTSR and the SUR USAR Chapter 14 analyses bound the actual plant setpoints. The changes in the safety analysis OTΔT and OPΔT setpoints impact the timing of the reactor protection system actuation. The acceptance criteria for the noted events remain the same, and are not affected by the change in OTΔT and OPΔT setpoints.

d. **Does this proposed TS change invalidate the USAR Chapter 14 analyses as reviewed by the staff in the RTSR for KNPP LAR 187?**

**NMC Response:**

Changing the $\Delta T_0$ is not a Technical Specification (TS) change (specific change discussed in #21a above). The USAR Chapter 14 analyses, as reviewed by NRC in the RTSR for KNPP LAR 187, were performed at assumed $\Delta T_0$ operating conditions that bound the expected plant conditions at the Stretch Uprate (SUR) reactor core power of 1772 MWt. Therefore these analyses are valid for SUR operation.

22. **The licensee proposes to change the TS 2.1.c safety limit for peak fuel centerline temperature from < 4700 °F to < 5080 °F (and decreasing by 58 °F per 10,000 MWD/MTU of burnup). Please provide a Reference to the Nuclear Regulatory Commission (NRC) approval (i.e., topical report safety evaluation report) for this proposed limit and provide the technical justification for its application to KNPP and Framatome fuel. Are any adjustments to the proposed safety limit necessary to account for burnable poisons? If so, provide the values for the adjustment necessary and the technical basis for the values.**

**NMC Response:**

The NRC approved the fuel centerline temperature limit as part of WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," S. L. Davidson, T. L. Ryan, April 1995, and WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," D. S. Huegel, J. D. Andrachek, C. E. Morgan, January 1999. The 5080°F minus 58°F/10,000 MWD/MTU fuel centerline melt limit has been the fuel centerline melt

limit for Westinghouse fuel since the mid-1960's. This limit is an industry accepted limit for Westinghouse fuel, backed by technical evaluations performed by Westinghouse. The Westinghouse data is applicable to all uranium dioxide fuel, so the fuel centerline limit is also applicable to the co-resident Framatome fuel remaining in the core. In WCAP-12610-P-A, this limit is also documented as applicable to ZIRLO™ clad fuel.

It is not necessary to adjust this limit to account for burnable poisons. Fuel with gadolinia poison has the fuel uranium enrichment reduced to ensure that the fuel with the gadolinia burnable poison satisfies all safety analysis acceptance criteria including the fuel centerline melt limit.

23. **The licensee proposes to change TS 2.3.a.3.A wording for f(I) from "An even function" to "A function" and states that this is an editorial change. Please discuss why the TS is currently worded as "an even function" and provide justification for the proposed wording change.**

**NMC Response:**

"An even function" is changed to "a function" for accuracy. According to "Advanced Engineering Mathematics" by C.R. Wylie, copyright 1960, a function f(t) is even when f(-t) = f(t). The f(delta-I) function is not an even function since f(-delta-I)does not equal f(+delta-I). Therefore, the word "even" is deleted.

24. **Commitment No. 1 of the licensee's submittal letter (Reference 1) states that an evaluation of the thermal and hydraulic safety analyses for the Framatome aircraft nuclear propulsion (ANP) fuel demonstrates that the departure from nucleate boiling ratio (DNBR) design basis is met for the Framatome fuel in Cycle 26, and the licensee commits to revise and update all documents for the stretch power uprate to address Framatome fuel DNBR design basis.**

a. **Please list the documents for which this commitment applies.**

**NMC Response:**

The documents to be revised are still being evaluated as part of the Stretch Power Uprate modification process, and are required to be revised as part of the pre-requisites of the power uprate implementation plan. As stated in our submittal, Commitment 1, the appropriate documents will be revised prior to implementation of the uprate.

b. **During the review of KNPP LAR No. 187 (Westinghouse 422 Vantage+ fuel transition), the staff requested that the licensee provide Framatome/ANP Non-loss-of-coolant accident (non-LOCA) transient and accident analyses discussions and results (Reference 2, Attachment 3 Request for Additional Information (RAI), Question 29). The licensee's response stated that for the current and MUR uprate power levels, adequate DNBR margin will exist for the Framatome/ANP fuel. However, to support the stretch power uprate, the licensee stated that the existing DNBR margin is not adequate to offset the power increase effects. In response to the staff's RAI, the licensee stated that the thermal-hydraulic analyses for the Framatome ANP fuel will**

**be generated during the reload safety evaluation process and will be documented in the Reload Safety Evaluation report and in the Stretch Power uprating submittal. Please provide this information.**

**NMC Response:**

In the KNPP response (Letter NRC-03-016 of 2/27/03, Attachment A, page 30) to your RAI question 29 to the KNPP LAR 187 submittal, we did not state, "... that the existing DNBR margin is not adequate to offset the power increase effects...", as stated in your question above. The RAI response states, "However, the analytical margin to the DNBR limit for the Framatone ANP fuel for the fuel transition and the full stretch uprate can be expected to decrease somewhat."

The Framatome ANP (FANP) thermal and hydraulic (T&H) analysis (DNBR analysis) results for cycle 26 SUR conditions will be documented in a Cycle 26 RSE supplement (this C26 RSE supplement will be applicable to SUR operation in C26). Given the FdH burndown (FdH for FANP fuel is< 1.38 after 9.5 GWD/MTU) after 9.5 GWD/MTU C26 (SUR will be implemented after 9.5 GWD/MTU C26), the FANP DNBR analysis for SUR is bounded by the existing C26 analysis performed for the Measurement Uncertainty Recapture (MUR) power uprate. The supplement to the C26 RSE is not yet completed. Per our Regulatory Commitment #1 (reference Attachment 7 of our submittal), the supplement to the RSE, and all associated documents, will be completed prior to implementation of the SUR.

25. **Reference 1, Attachment 4, Section 2 discusses the NSSS parameter values used for the power uprate analysis.**

    a. **In determining the NSSS parameter values, the licensee considers Westinghouse 422V+ fuel only, and states that it is not appropriate to consider any transition core effects. Because the upcoming cycle consists of the first 422V+ transition core, please provide the technical basis for this statement.**

    **NMC Response:**

    Since the Westinghouse fuel design has a greater thermal hydraulic resistance and pressure drop than the Framatome fuel design, it is conservative to model the Westinghouse fuel design only in the Performance Capability Working Group (PCWG) design analyses. Modeling only the Westinghouse fuel design yields more conservative NSSS design parameters, e.g. higher core delta T and higher core outlet temperature, for use in the stretch uprate NSSS safety/thermal analyses and evaluations. The PCWG analyses conservatively assume thermal design flow for reactor core flow. Thermal design flow sufficiently bounds the actual, mixed core flow for the fuel transition core designs.

    b. **Table 2.1-1 provides a listing of the NSSS parameter values used in the power uprate analyses. The values used appear to be consistent with those used in the 422V+ fuel transition amendment (RTSR). However, the**

**licensee states in Section 2.1.2 that the TAVG range was narrowed slightly from the previous range, and the NSSS and reactor power are not bounded by the values used in the RTSR analyses. Please clarify this inconsistency and verify that the USAR Chapter 14 transients and accidents as reviewed by the staff for the RTSR remain bounding for the stretch power uprate.**

**NMC Response:**

The statement in Reference 1, Attachment 4 of our submittal, Section 2.1.2 is confusing because there are two Table 1-2's in Attachment 4 to LAR 187 (fuel transition amendment). One (page 4-13 of Attachment 4 to LAR 187) is for the Replacement Steam Generator Program (PCWG-2534) at a NSSS power of 1657.1 MWt, and the second one (page 4-14 of Attachment 4 to LAR 187) is for the RTSR/Uprate Program (PCWG-2707) at a NSSS power of 1780 MWt.

The USAR Chapter 14 transient analyses reviewed by the NRC for the Reload Transition Safety Report (RTSR) (LAR 187) were for an NSSS power of 1780 MWt, and remain bounding for the stretch uprate (SUR) conditions. The statement in section 2.1.2 of attachment 4 of LAR 195 states " The only NSSS parameters in Table 2.1-1 that are not bounded by those from the RTSR/RSG Programs are the NSSS and reactor power, feedwater temperature, and steam flow". This section could have been more explicit in differentiating between the two separate tables with the same number (Table 1-2), and which program parameters are bounding.

The NSSS and reactor power, feedwater temperature, and steam flow parameter values in Table 2.1-1 of attachment 4 of our submittal are the same as the values in Table 1-2 (page 4-14) of attachment 4 to LAR 187 for the RTSR/Uprate Program (PCWG-2707). The values in Table 1-2 (page 4-14) for these parameters bound those of Table 1-2 (page 4-13) of attachment 4 to LAR 187. The RTSR and the uprate programs used consistent NSSS parameter values based on the power uprate Performance Capability Working Group (PCWG) analyses for a NSSS power of 1780 MWt.

26. **Reference 1, Attachment 4, Section 5.2.5.3, addresses core bypass flow. The licensee determined that the current core bypass flow limit of 7.0 percent of total vessel flow can be maintained at the uprated power conditions. Please discuss the methodology used to reach this conclusion.**

**NMC Response:**

The current core bypass flow limit of 7.0 percent of total vessel flow is maintained for stretch uprate (SUR) conditions. The 7.0% core bypass flow, which is a bounding design core bypass flow value, is based on reactor coolant pump design, reactor coolant system flow, reactor vessel design, and includes consideration of the core not having fuel thimble plug inserts. Since these components/parameters have not changed for the power uprate, the 7.0% core bypass flow design limit remains applicable to the stretch uprate (SUR) conditions.

**27. Reference 1, Attachment 4, Section 5.3, "Fuel Assemblies" addresses structural integrity of the Westinghouse 422V+ fuel assemblies and concludes that adequate grid load margin exists such that core coolable geometry and control rod insertion requirements are satisfied. The staff requested additional information regarding this aspect as part of the fuel transition amendment request (Reference 2, Attachment 3 RAI's, Question 13) and the issue was resolved in the KNPP letter to the staff dated April 2, 2003 (Letter No. NRC-03-037). Are the conclusions reached in Section 5.3 based on the same analyses and consistent with those described in NMC Letter No. NRC-03-037? Please provide clarification.**

**NMC Response:**

The conclusions reached in section 5.3 of attachment 4 to our stretch power uprate submittal are based on the same analyses and are consistent with the conclusions that were performed for the fuel transition submittal (reference KNPP Letter #NRC-02-067 of 7/26/02) and documented in NMC letter NRC-03-037. The conclusion reached in section 5.3, that adequate grid load margin exists such that core coolable geometry and control rod insertion requirements are satisfied, applies to the fuel assemblies operating at SUR.

**28. Reference 1, Attachment 4, Section 6.2.2, discusses the anticipated transients without scram (ATWS) analyses performed to support the stretch power uprate.**

**a. Please discuss which power levels and moderator temperature coefficient (MTC) values were analyzed, and which produced the limiting results for both the reactor coolant system (RCS) pressure and SG pressure cases.**

**NMC Response:**

The ATWS analysis, that supports the stretch power uprate, is based on full power conditions (1780 MWT NSSS power) and a moderator temperature coefficient (MTC) of -8 pcm/°F. These conditions are consistent with the basis for the ATWS rule (ATWS Final Rule – Code of Federal Regulations 10 CFR 50.62 and Supplementary Information Package, Reduction of Risk from Anticipated Transient Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants). *Since only one accident was analyzed at a given set of conditions, it is* assumed to be limiting for both the RCS and S/G pressure cases (see NRC #28d below for limiting accident).

**b. Please list the analytical codes and methods used for the ATWS analyses. Provide technical justification if not consistent with those listed in KNPP USAR Section 14.1.11.**

**NMC Response:**

The LOFTRAN computer code was used to perform the KNPP ATWS analysis for the stretch power uprate, consistent with the analysis basis for the ATWS Final Rule. The use of LOFTRAN is also consistent with the analysis basis for the KNPP USAR Section 14.1.11.

c. ***Please discuss the analysis assumptions with regard to the physical plant configuration. For example, did the analyses consider SG tube plugging, and what levels? How many power-operated relief valves were assumed to be available? What level of AFW was assumed available? Was any control rod insertion credited? Provide this discussion for both the RCS Pressure and SG Pressure cases.***

**NMC Response:**

The ATWS analysis assumptions with regard to the physical KNPP configuration are as follows:

a. A value of 0% steam generator tube plugging (SGTP) is assumed, consistent with the analysis basis (Westinghouse Letter NS-TMA-2182, "Anticipated Transient Without Scram for Westinghouse Plants," December 1979) for the ATWS Final Rule. The value of 0% SGTP is assumed since this leads to more bounding results in the limiting RCS pressure ATWS events.

b. Operation of two pressurizer PORVs is assumed, consistent with the analysis basis for the ATWS Final Rule.

c. The auxiliary feedwater flow rate, corresponding to 3 AFW pumps running, is based on the SG pressure. The flow rates used in the analysis are as follows:

| AFW Flow (gpm) | SG Press (psia) |
|---|---|
| 0. | 1527.5 |
| 60. | 1485.8 |
| 120. | 1414.4 |
| 180. | 1313.2 |
| 240. | 1182.2 |
| 300. | 1021.5 |
| 360. | 831.0 |
| 420. | 610.8 |
| 480. | 360.8 |

d. Control rod insertion was not assumed.

e. The assumptions discussed above were applied to both the RCS pressure and MS pressure cases.

d. **It appears that the licensee reanalyzed the loss of normal feedwater flow ATWS event only. Is this the limiting ATWS event for KNPP for both RCS pressure and SG pressure?**

**NMC Response:**

According to the generic ATWS analysis supporting Westinghouse Letter NS-TMA-2182, "Anticipated Transient Without Scram for Westinghouse Plants," December 1979, the Loss of Normal Feedwater (LONF) is the limiting ATWS transient for a 2-loop plant design, and is limiting with respect to RCS pressure. Secondary system overpressure (SG pressure) is not an ATWS event criteria. Therefore, the LONF event was analyzed for the KNPP stretch power uprate.

The Loss of Load (LOL) ATWS is analyzed for Westinghouse plants with steam-driven main feedwater (MFW) pumps, since an initiating turbine trip event causes a loss of condenser vacuum with consequential loss of main feedwater. For plants with motor-driven MFW pumps, a loss of load event does not result in the loss of MFW, as the pumps would continue to run. Since Kewaunee has two motor-driven MFW pumps, the LOL ATWS event was not analyzed for the KNPP stretch power uprate.

e. **The licensee's core operating limit report (COLR) contains a requirement that, "The reactor will have a MTC no less negative than -8 pcm/_F for 95 percent of the cycle time at full power." Please discuss the administrative controls in place that ensure this operational requirement is satisfied. Also, what controls are in place to ensure this requirement in the COLR is not changed and the basis for the ATWS rule is preserved?**

**NMC Response:**

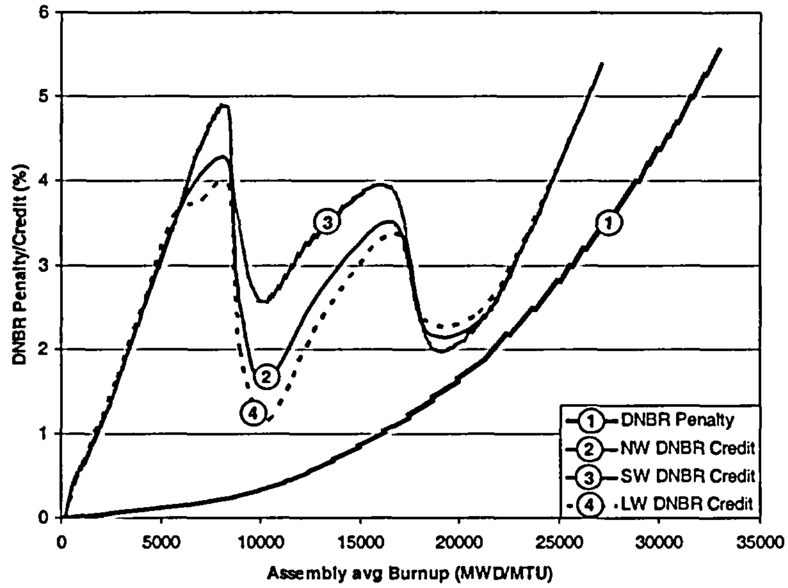
The COLR requirement that "The reactor will have a moderator temperature coefficient no less negative than -8 pcm/DegF for 95% of the cycle time at full power." is a reload safety evaluation criterion that must be verified formally through appropriately conservative design calculations. It is not an operational requirement since it is impractical to measure HFP MTC during operation. Low Power Physics Testing is performed prior to startup to verify the design calculations and conservatisms. Monthly power distribution surveillances provide ongoing confirmation of the design calculations. Administrative controls in place to ensure this requirement is satisfied are that changes to the MTC requirement in the COLR are only allowed when the underlying ATWS safety analyses performed using approved methods support the change. A change to the ATWS MTC requirement would require a change to the KNPP approved ATWS method of analysis. The ATWS analysis methodology change would require use of the 50.59 process which would likely need (this is determined by application of the 50.59 process) NRC approval for the methodology change. Therefore, the basis for the ATWS rule, as applied to KNPP, is preserved through existing change controls.

29. **Reference 1, Attachment 4, Section 7.1 discusses the core thermal-hydraulic design. To accommodate the stretch power uprate conditions, the licensee modified certain aspects of the DNBR margin calculations.**

a. **The RTSR analyses included a 2.6 percent rod bow DNBR penalty. For the stretch uprate, the licensee reduces the rod bow penalty from 2.6 percent to 0 percent for Cycle 26 by evaluating the F H burndown DNBR credit during the entire cycle. Please provide the technical justification for this change in the rod bow penalty. Include descriptions of the F H burndown DNBR credit and the analyses performed to quantify F H burndown DNBR credit, and provide a reference to the NRC-approved methodology applied. Also, provide the results of the analyses which demonstrate that the necessary rod bow penalty is offset by the F H burndown DNBR credit for Cycle 26.**

**NMC Response:**

The 2.6% rod bow DNBR penalty is based on an analysis performed in Refs. 1 and 2 where it was demonstrated that credit for FdH burdown can be used after 24,000 MWD/MTU (assembly average burnup) to limit the rod bow DNBR penalty to 2.6 % for 0.422 OD diameter fuel and WRB-1 DNB correlation. Beyond 24,000 MWD/MTU the rod bow penalty increases beyond 2.6% but it was shown that for a representative set of cycles using Westinghouse fuel, the FdH burdown conservatively offset the additional DNBR penalty and therefore a 2.6% rod bow DNBR penalty is acceptable. This is the general approach that does not require any cycle specific analysis. The method used to evaluate the rod bow DNBR penalty for Kewaunee Cycle 26 was as analogous to the method used for Ref. 1. However, the analysis was performed using the peak FdH as a function of assembly burnup for the specific cycle only (long, normal and short EOC 25 windows). BOL conditions set the reference for the rod bow DNBR penalty (0%) and the FdH burdown credit (0%). The rod bow DNBR penalty increase with assembly average burnup was calculated according to the approved methodology described in Ref. 1. The peak FdH as a function of assembly burnup was calculated by ANC for Kewaunee Cycle 26 (long, normal and short windows). The credit/penalty curves due to FdH variation were calculated based on a conservative one-to-one relation with DNBR (this is much less than the minimum DNBR sensitivity to FdH, which is calculated to be 1.898 for the set of RTDP conditions used in the Design Limit DNBR calculation). The result of the comparison is shown below.




The figure shows that the rod bow DNBR penalty is offset by a conservative FdH burdown credit over the entire life of Kewaunee Cycle 26 core. Therefore the rod bow DNBR penalty was evaluated to be 0% for Kewaunee cycle 26. Please note that even though a similar methodology as the one used in the approved topical report (Ref. 1) was used, since the analysis was performed based on Kewaunee Cycle 26 peak FdH only, this is in non conformance with the current approved methodology and therefore needs a specific approval for use during this cycle.

Please also note that this analysis was performed to address the low DNBR margin (0.3%) that was available for the first transition cycle at uprate conditions so as to increase the net DNBR margin that could be used for unexpected events. With the second and third transition cycles, the transition core DNBR penalty will decrease and therefore the total DNBR margin will return to a more comfortable level without the need of a cycle specific rod bow DNBR penalty evaluation.

Reference 1: WCAP-8691, Rev. 1, "Fuel Rod Bow Evaluation."
Reference 2: "Request for Reduction in Fuel Assembly Burnup Limit for Calculation of Maximum Rod Bow Penalty," Letter, C. Berlinger (NRC) to E. P. Rahe, Jr. (Westinghouse), June 18, 1986.

b. **The staff currently accepts that rod bow penalty be limited to fuel burnup of 24,000 MWD/MTU because of burndown effects. Is the approach being applied to eliminate or offset the 2.6 percent rod bow penalty, in essence, double accounting for this F H burndown DNBR credit? Please clarify.**

**NMC Response:**

As described in the answer to question 29a above, the reduction of the rod bow DNBR penalty to 0% for Kewaunee Cycle 26 is demonstrated over the entire life

of the C26 core independent of the approved 24,000 MWD/MTU limit. Therefore, the FdH burndown credit is accounted for only once.

c. **Does the COLR need to be revised to reflect any changes in F H?**

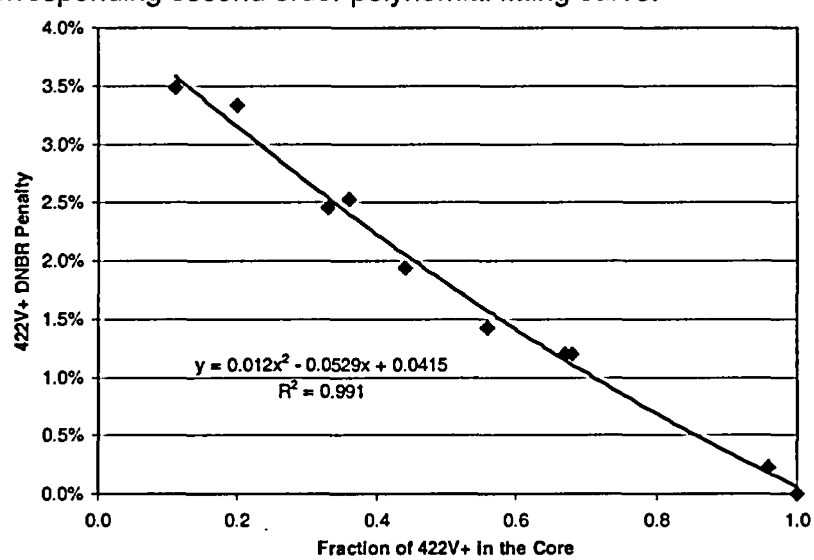
**NMC Response:**

The KNPP COLR does not need to be revised for the Westinghouse fuel design FdH limit. However, the cycle 26 COLR will be revised for SUR to reflect a reduced Framatone ANP fuel FdH limit.

d. **The licensee calculated a transition core penalty of 2.5 percent for Cycle 26. Figure 4-5 of the licensee's RTSR submittal (Attachment 4, Westinghouse report) provided a figure of transition core penalty as a function of the amount of 422V+ fuel loaded in the core. RAI No. 24 of the RTSR LAR No. 187 questioned the linear relationship of this figure, and in their response, the licensee stated that an additional penalty was taken to account for variance in the curve fitting and that the results will be presented in the stretch uprate submittal. Please provide a corresponding figure to Figure 4-5 of the RTSR submittal which incorporates the additional transition core configurations evaluated for the stretch power uprate to justify a 2.5 percent transition core penalty. Provide the fraction of 422V+ fuel assemblies to be loaded in the KNPP core for each transition operating cycle.**

**NMC Response:**

The transition core DNBR penalty as a function of the fraction of 422V+ assemblies in the core was calculated for additional transition core patterns. The result of the analysis is presented in the following figure along with the corresponding second order polynomial fitting curve.



In order to account for the variance in the curve fitting 0.26 % DNBR penalty is added:

$$\text{DNBR penalty (\%)} = 1.20x^2 - 5.29x + 4.41$$

where x is the fraction of 422V+ assemblies loaded in the core

The first transition core for Kewaunee Cycle 26 will have 48 422V+ assemblies (44 feed + 4 lead), therefore the fraction of 422V+ assemblies loaded in the core is 48/121=0.3967, and the DNBR transition core penalty is 2.5%.

The fraction of 422 V+ fuel assemblies in each of the transition cycles and the corresponding DNBR transition core penalty are shown in the table below.

| Transition cycle | $1^{st}$ (x=0.3967) | $2^{nd}$ (assume x=2/3) | $3^{rd}$ (assume x=1) |
|---|---|---|---|
| DNBR penalty | 2.78 % | 1.42 % | 0 % |

e. **The licensee is reducing the design limit DNBR value from 1.24 to 1.23 by taking into account the latest calculated instrumentation uncertainties, and references WCAP-15591, Revision 1. This same WCAP and revision were used for the RTSR (LAR No. 187) analyses to calculate a 1.24 design limit. Please provide clarification regarding the latest uncertainties and that these latest uncertainties were not already credited in the RTSR limit of 1.24.**

**NMC Response:**

The Design Limit DNBR of 1.24 was calculated using a set of conservative instrumentation uncertainties that were believed to bound the actual instrumentation uncertainties. At the time this Design Limit DNBR was calculated, the actual instrumentation uncertainties were not yet calculated. The basis for the validity of this set of conservative uncertainties, i.e., that the set of conservative uncertainties do in fact bound the actual calculated uncertainties, is WCAP-15591, Rev. 1. The Design Limit DNBR of 1.23 was later calculated using calculated instrumentation uncertainties from WCAP-15591, Rev. 1 and was subsequently used in the DNBR analysis. The reference to justify both sets of instrumentation uncertainties is therefore the same. It is understood that a new Design Limit DNBR should be evaluated in the event the actual calculated uncertainties need to be increased.

f. **Note 1 of Table 7.1-2 states that enough DNBR margin was retained to cover rod bow, instrumentation bias and transition core penalties for the W-3 DNBR correlation. Please provide a corresponding DNBR margin summary table for the W-3 correlation.**

**NMC Response:**

The W-3 DNB correlation is used when WRB-1 is not applicable (Hot Zero Power (HZP) SteamLine Break (SLB) and Feedwater Malfunction (FM) events and below the first mixing vane grid for Rod Withdrawal from Subcritical (RWFS) event). The DNBR margin summary table for W-3 is provided below. Note that STDP was used to analyze these events and that W-3 has different correlation limit depending on the range of system pressures.

| Parameter | Non-RTDP events / W-3 | | |
|---|---|---|---|
| | RWFS | HZP SLB | HZP FM |
| DNBR Correlation | W-3[1] | W-3 | W-3 |
| DNBR Correlation Limit | 1.30 | 1.45 | 1.45 |
| DNBR Safety Analysis Limit | 1.416[2] | 1.580[2] | 1.580[2] |
| DNBR Margin | 8.21 % | 8.21 % | 8.21 % |
| Instrumentation Bias Penalty | 0 %[3] | 0 %[3] | 0 %[3] |
| Rod Bow DNBR Penalty | 0 %[4] | 0 %[5] | 0 %[4] |
| Transition Core DNBR Penalty | -2.50 %[6] | -2.50 %[6] | -2.50 %[6] |
| **Available DNBR Margin** | **5.71 %** | **5.71 %** | **5.71 %** |

[1] W-3 is used below the first mixing vane grid for the RWFS event.

[2] W-3 Safety Analysis Limit DNBR is arbitrarily set such as to maintain the generic 8.21 % DNBR margin existing for RTDP. This limit is back calculated using the following equation: (DNBR margin) = 1-(DNBR Correlation Limit)/(DNBR Safety Analysis Limit).

[3] For non-RTDP event, instrumentation bias effects are included in the transient analysis.

[4] The W-3 rod bow DNBR penalty for RWFS and HZP FM events was calculated to be 0%.

[5] For HZP SLB there is no rod bow DNBR penalty applicable because of the low pressure.

[6] Transition core DNBR penalty calculated using WRB-1 correlation is also used for W-3. This is a bounding approach because 1) WRB-1 DNBR penalty bounds all plant conditions and 2) W-3 is not more sensitive than WRB-1 to parameter change (such as mass flux for instance).

30. **Reference 1, Attachment 4, Section 7.3 discusses fuel rod design and performance. In this section, the licensee states that pending approval of Addendum 1 to WCAP-10125 by the NRC, subsequent re-evaluation of the stress values will be performed to confirm the proposed clad stress criterion is met. Please verify that this evaluation was performed and documented in a letter dated March 21, 2003 (letter no. NRC-03-032), and that the conclusions remain valid for the stretch power uprate conditions.**

**NMC Response:**

In a letter (#NRC-03-032) to the NRC dated March 21, 2003, the NMC confirmed that these evaluations had been completed acceptably. The conclusions on cladding stress values meeting cladding stress acceptance criteria, documented in letter #NRC-03-032, remain valid for the SUR operating conditions.

31. **In Table 5.1-3 of the submittal the peak vessel fluence for 33 effective full-power years of operation are listed as $3.56 \times 10^{19}$ n/cm$^2$ vs $3.34 \times 10^{19}$ n/cm$^2$ in WCAP-14279 Rev. 1. Apparently, the $3.34 \times 10^{19}$ n/cm2 was derived from the $3.49 \times 10^{19}$ n/cm$^2$ value in WCAP- 14279. It seems that the $3.56 \times 10^{19}$ n/cm$^2$ value was derived from WCAP-14279 Rev. 1 by rationing for the uprate. However, the original value seems to have been derived using the FERRET code which has not been approved. Please justify the use of the $3.56 \times 10^{19}$ n/cm$^2$ value.**

**NMC Response:**

According to the Pressure Temperature (PT) Curve report (WCAP-14278, Rev. 1, Kewaunee Heatup and Cooldown Limit Curves for Normal Operation, page 4-1) calculated fluence projections, not the FERRET code, were used to develop the PT Curves. The calculated fluence used was $3.34 \times 10^{19}$ n/cm$^2$ @ 33 EFPY.

The fluence value of $3.56 \times 10^{19}$ n/cm$^2$ at 33 EFPY was not derived from WCAP-14279-Rev.1. This fluence value is calculated using the guidance of Reg. Guide 1.190. The FERRET code was not used to derive this calculated fluence value. Therefore the use of $3.56 \times 10^{19}$ n/cm$^2$ at 33 EFPY is justified.

KNPP Letter #NRC-03-047 of 4/30/03, Responses to RAIs Regarding LAR 193, MUR Power Uprate, response to Question #4 (page 30 of Attachment 1) provides further information that is pertinent to this subject.

32. **Please provide copies of References 14 – 17, Page 6-8 (WCAP-8339, WCAP-8471-P-A, WCAP-8471-A, NSID-TB-86-08, and CLC-NS-309)**

**NMC Response:**

Except as noted below, the requested documents are enclosed as enclosures B and C. Please note that you ask for both WCAP-8471-P-A and WCAP-8471-A. Reference 15 on page 6-8 of Attachment 4 of our submittal is WCAP-8471-P-A and WCAP-8472-A. WCAP-8472 is the non-proprietary version of WCAP-8471-P as stated in the front of

WCAP-8471-P. Both of these documents have been previously provided to, and approved by the NRC. Therefore, these WCAPs are not included with this submittal. A copy of CLC-NS-309 has been previously provided.

**33. Please provide justification for assuming a saturation pressure of 35 psia in boric acid accumulation calculations in light of the anticipated behavior of long-term containment pressure.**

**NMC Response:**

The 35 psia assumption referred to is not really a containment pressure assumption. Rather it is a recognition that at 35 psia RCS backpressure, the low head pumps will be injecting into the upper plenum. The calculational scenario for the analysis is a buildup of boric acid due to the absence of flow to the upper plenum. If the low head pumps are injecting into the upper plenum, core flushing flow is provided. The Emergency Operating Procedures (EOP) direct the operator to establish low-head recirculation (via the upper head injection lines) at RCS pressures below 150 psig, and confirm a minimum flow (1500 gpm). Therefore, basing the boric acid solubility limit on a pressure well below 150 psig is justified.

**34. Is it possible for some small-break LOCA sequences to cause a loss of natural circulation in the RCS for an extended time so that boric acid is accumulating in the core? Please address the question from the viewpoint of two cases: (1) with an RCS pressure higher than the residual heat removal (RHR) shutoff head pressure; and (2) with an RCS pressure lower than the RHR shutoff head pressure.**

**NMC Response:**

For the scenario where the RCS can be refilled, natural circulation will be established and boron precipitation cannot occur.

For the scenario where the RCS cannot be refilled, natural circulation is not credited. When the RCS pressure falls sufficiently below the low head cut-in pressure (150 psig per EOPs), boron precipitation is precluded by core flushing flow provided by the upper plenum injection. Calculations performed as part of the uprate program demonstrate that even if the upper plenum injection is established as late as 18 hours after the LOCA, the vessel boron concentration will still be 4 weight % under the boron precipitation point, assuming a saturation pressure of 35 psia.

If the RCS pressure remains above the RHR cut-in pressure for an extended period of time (>18 hours), the potential for boron precipitation is not of concern since:

- At higher RCS pressures, the boric acid solubility limit is significantly higher than that assumed in the uprate analyses.
- At higher RCS pressures, the core boil-off would be less than that calculated in the uprate analyses due to the greater subcooling of the injected SI.
- Core boil-off for this break would be significantly less than that calculated in the uprate analyses since a significant amount of heat would be removed by the SGs (via steam dump/auxiliary feedwater).

Note that this scenario is highly unlikely since it applies only for a narrow range of small break sizes that are so large that the RCS cannot be refilled, and so small that the RCS cannot be depressurized after an extended period of time.

**35. This question was deleted since it was a repeat of Question Number 6 regarding Station Blackout.**

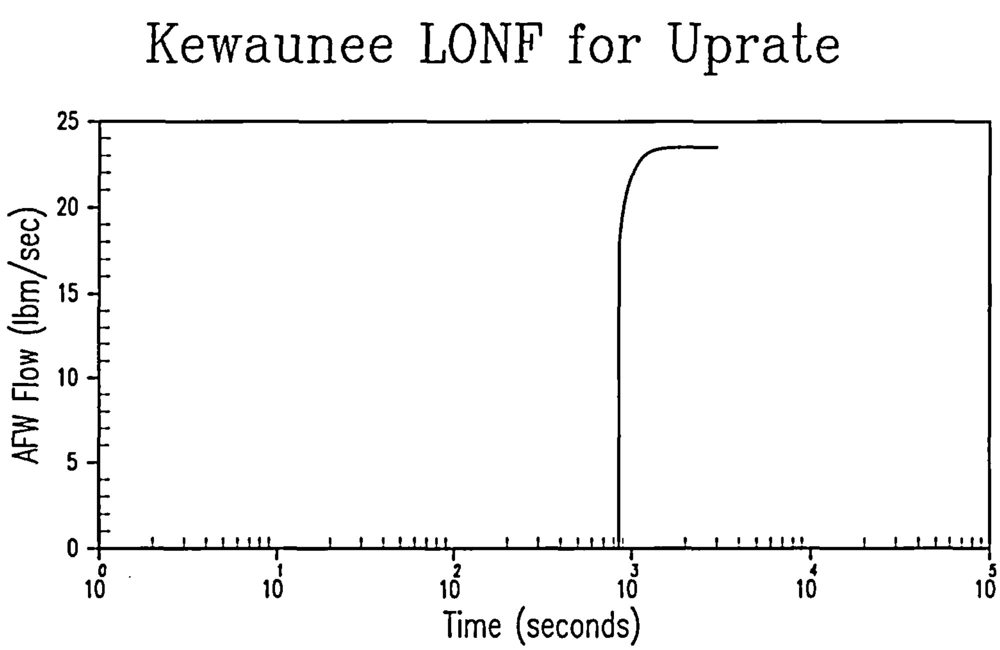
**NMC Response:**

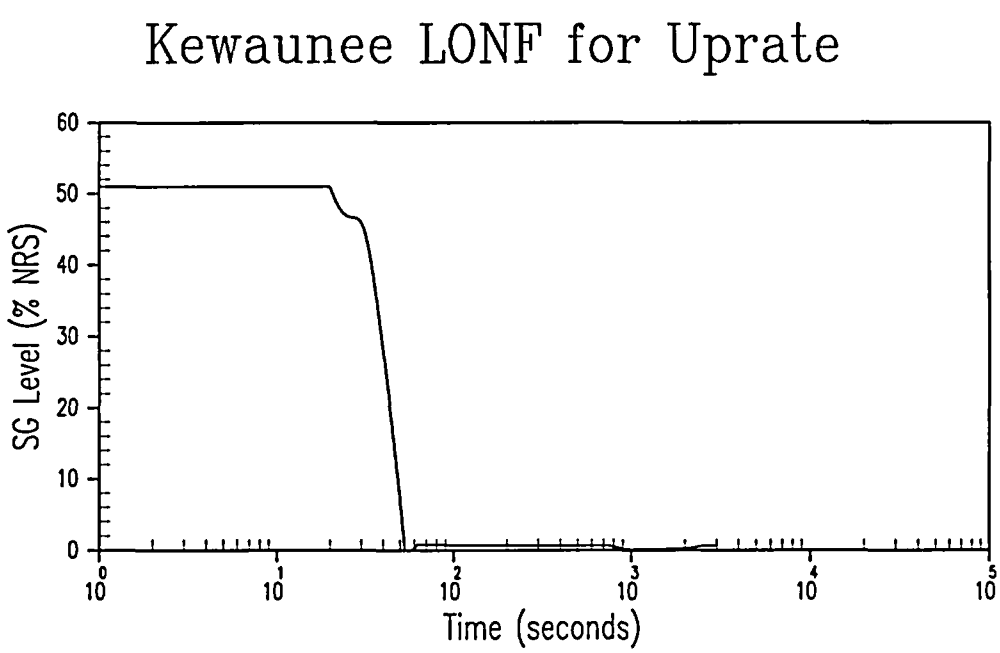
None Required.

**36. To support the results of the loss of normal feedwater transient, please provide the following:**

**a. Transient curves of AFW flow rate and SG water level.**

**NMC Response:**






b. **Discuss the need for time delay of AFW flow to SGs while the plant is operated below 15 percent rated power.**

**NMC Response:**

TS 3.4.b.6 states (note TS 3.4.b.6 is not being changed as part of our submittal): When the reactor is <15% of RATED POWER, any of the following conditions may exist without declaring the corresponding auxiliary feedwater train inoperable:

A. The AFW pump control switches located in the control room may be placed in the "pull out " position
B. Valves AFW 2A AFW 2B may be in a throttled or closed position
C. Valves AFW 10A and AFW 10B may be in the closed position

The AFW system time delay assumption of 630 seconds is used for the LONF transient analysis cases initiated from 15% of Rated Power (RP), or less. This time delay analysis assumption is needed at or below 15% RP to demonstrate that operators have at least 10 minutes to manually initiate AFW, and to align the system for injection into the steam generators given the AFW conditions allowed by TS 3.4.b.6.

c. **The results of a loss of normal feedwater transient which maximizes the RCS peak pressure.**

**NMC Response:**

With respect to peak reactor coolant system pressure, the loss of normal feedwater (LONF) transient analysis is bounded by the analysis of the loss of load / turbine trip (LOL/TT) transient. This is because the LOL/TT transient results in a more limiting power mismatch between the primary and secondary sides. In the LOL/TT analysis, following the loss of secondary load (and assuming a coincident feedwater isolation), the primary-side operates at full power (102%) until rod motion occurs approximately eight to ten seconds into the transient. This contrasts with the LONF analysis in which the turbine trip occurs coincident with the reactor trip (within 1 second). Having a longer period of primary-to-secondary power mismatch is conservative with respect to primary-side pressurization. Based on this, a LONF case that maximizes the peak RCS pressure is not explicitly analyzed.

d. **The results of a loss of normal feedwater transient assuming that the AFW flow is delivered within one minute following the event to show the effect of overcooling at the beginning of the transient.**

**NMC Response:**

With respect to overcooling, the LONF transient analysis is bounded by the analysis of the feedwater system malfunction (FWM) transient in which a main feedwater flow increase of 150% of nominal is postulated. As main feedwater flow is significantly greater than AFW flow, the cooldown associated with a FWM transient is more severe than a cooldown associated with a LONF transient. Based on this, a LONF case that maximizes the cooldown at the beginning of the transient is not explicitly analyzed.

e. **Discuss the provisions made in plant emergency operating procedures (EOPs) for controlling AFW at the beginning of the event to prevent excess cooldown during this event.**

**NMC Response:**

In the event of a Reactor Trip with Safety Injection: Procedure E-0, Reactor Trip or Safety Injection, will stop the TDAFWP if both Motor Driven AFW Pumps are running, and will stop the other MDAFWP and control flow if a cooldown is occurring.

In the event of a Reactor Trip with no Safety Injection: Procedure ES-0.1, Reactor Trip Response, will stop the TDAFWP if at least one MDAFWP is running and control flow if a cooldown is occurring. ES-0.1 is entered from E-0.

f. **Discuss the mechanism of turning around the peak RCS pressure prior to the AFW flow delivering to SGs.**

**NMC Response:**

Pressurizer spray actuation provides the means for controlling the pressurizer pressure prior to AFW flow delivery. The pressurizer sprays are assumed to operate because it is conservative with respect to the LONF analysis objective of maximizing the pressurizer mixture volume.

g. **Since a loss of normal feedwater transient require AFW flow from two pumps, explain why other heat up transients and/or a SBLOCA are not effected.**

**NMC Response:**

Other heat up transients and/or the SBLOCA accident are not limiting with respect to AFW flow assumptions. Therefore, these other transients and accidents continue to be analyzed with the assumption of only one AFW train delivering flow to the SG's, and continue to be acceptable with the one pump AFW flow assumption at SUR conditions (reference the response to SUR RAI #20). Only the LONF accident at SUR conditions requires AFW flow from two AFW pumps to mitigate the consequences of the transient to assure that the safety analysis acceptance criteria are met. The LONF accident assumes continued heat input from the Reactor Coolant Pumps (RCPs), while the other heat up accidents do not.

37. **Provide the basis for assuming an initial pressurizer water level at 48 percent of span. Do TS at KNPP support this assumption?**

**NMC Response:**

The initial pressurizer level assumption of 53% in the LONF transient analysis is based on the full power nominal pressurizer level of 48% plus the calculated pressurizer level uncertainty of 5%. A higher initial pressurizer level is conservative for the LONF analysis, as it minimizes the margin to filling the pressurizer water-solid. There is no specific Technical Specification delineating a pressurizer level control setpoint or band at KNPP. However, the pressurizer programmed level at KNPP is controlled at less than 48% level.

38. **The proposed TS 3.4-3 will permit the following changes to the AFW system when plant is operated at a power level below 15 percent of rated power: (a) AFW pump control switches located in the control room may be placed in the "pull out" position, (b) valves AFW-2A and AFW-2B may be in a throttled or closed position, and (c) valves AFW-10A and AFW-10B may be in the closed position. Please provide a discussion on this plant operational configuration relative to compliance with the ATWS rule in 10 CFR 50.62.**

**NMC Response:**

As a point of clarification, the Technical Specification (TS) change referred to in this question (on page TS 3.4-2) is new TS 3.4.b.7. However, this is not a new TS. It is just being re-numbered due to other changes. New TS 3.4.b.7 is the same as current TS

3.4.b.6 (page TS 3.4-2, Amendment #167). Under the current license, the AFW System automatic operation can be disabled <15% of rated power. As this is part of the current license, the AFW configuration in question has received prior regulatory approval regarding compliance with the ATWS Rule in 10 CFR 50.62.

39. **The proposed change to the TS basis (on page TS B3.4-3) indicates that a main steamline break (MSLB) accident as well as a loss of normal feedwater transient at 1772 MWt would require AFW flow from two AFW pumps. Please explain the reason why the existing MSLB analysis (assuming only one AFW pump feeding SGs) are still valid at KNPP at the power uprate conditions.**

**NMC Response:**

Both the existing main steam line break (MSLB) core response and containment integrity analyses conservatively assume maximum AFW system operation in order to maximize the severity of the RCS Cooldown, and for the containment integrity analysis, to maximize the source for mass and energy addition to containment. The existing MSLB core response analysis assumes bounding maximum AFW system operation equivalent to 1200 gpm of total flow at 35°F, corresponding to a conservative runout flow of 400 gpm per pump and all three pumps operating, initiated coincident with the main steam pipe rupture. The MSLB containment integrity analysis assumes maximum AFW system operation, based on conservative head-capacity curves with a runout flow of 400 gpm per pump and all three pumps operating, and that AFW flow is initiated conservatively soon after the pipe rupture. Therefore, the existing MSLB analysis of record bounds all modes of degraded or out-of-service AFW system components at the uprated 1772 MWt power level.

40. **Provide the results of a steam generator tube rupture (SGTR) thermal-hydraulic analysis using the event scenario consistent with EOPs. Assuming concurrent loss of offsite power and a stuck atmospheric dump valve (ADV) at the failed SG. Provide transient curves of primary and secondary system pressures and temperature, AFW flow rate, SG water levels, primary leak-rate, steam release rate from the SG safety valves and ADVs, etc. This information is needed to confirm that the estimated release of contaminated steam is conservative for the staff assessment of the radiological consequences.**

**NMC Response:**

The event scenario presented, which includes the assumed failure of an ADV, is outside the plant's design basis. This single active failure in the SGTR event is not required based on historical KNPP licensing analysis bases. Also, at KNPP, the ADVs are downstream of the Main Steam Isolation Valves (MSIV), and any steam relief through an ADV will be terminated when the MSIVs are closed.

The current SGTR analysis described in our submittal, Attachment 4, Sections 6.3 and 6.7.5, assumes a continual release of steam from the SG with the ruptured tube for a period of 30 minutes, assuming worst case conditions to maximize the steam release. After 30 minutes, it is assumed the Operators have completed actions to terminate the break flow and steam release. For the radiological consequences, the accident analysis assumes a 1% fuel defect, maximum iodine spiking in the RCS using a greater-than-

required spiking factor, and a secondary coolant iodine activity concentration at the Technical Specification limit. All of these assumptions are made to provide a conservative value for the estimated release.

**41. Provide the results of a SGTR thermal-hydraulic analysis to demonstrate that the SG will not be overfilled by AFW flow during this event.**

**NMC Response:**

The event scenario requested is outside the plant's design basis and therefore thermal hydraulic analysis results to demonstrate that the SG will not be overfilled by AFW are not presented.

The current SGTR design basis analysis, described in Attachment 4 of our submittal, Sections 6.3 and 6.7.5, is summarized in the response to RAI #40 above.

**42. Please provide a tabulation of all computer codes and methodologies used in the re-analyses including staff approval status, conditions and limitations, and how the conditions and limitations are satisfied for application at KNPP.**

**NMC Response:**

A tabulation of the computer codes and methodologies used in the re-analyses including the staff approval status, conditions and limitations and how they are satisfied for application to KNPP is provided in the tables below (copied from previous response [Letter #NRC-03-016 of 2/27/03] to NRC RAI #35 from LAR 187 submittal for fuel transition).

The following additional analyses were performed for SUR:

a) High Energy Line Break (HELB) (outside containment) mass and energy release analysis. This analysis uses LOFTRAN, and LOFTRAN is included below.
b) Containment integrity analyses. These analyses use GOTHIC 7 approved for application to KNPP (Letter from Anthony McMurtray to Thomas Coutu dated September 29, 2003, issuing amendment TAC # MB6408 and safety evaluation for GOTHIC 7 with MDLM).
c) Anticipated Transients without scram (ATWS) analysis uses RETRAN, and RETRAN is included below.
d) Loss of Normal Feedwater (LONF) analysis uses RETRAN, and RETRAN is included below.
e) Radiological accident analyses use alternate source term methods approved for application to KNPP (Letter from John G. Lamb (NRC) to Mr. Thomas Coutu (NMC), "Kewaunee Nuclear Power Plant – Issuance of Amendment Regarding Implementation of Alternate Source Term," dated March 17, 2003, TAC # MB4596).
f) Main Steam Line Break (MSLB)- containment response mass and energy release use methods approved by NRC (NMC topical report WPSRSEM P-A Revision 3).
g) Framatome ANP fuel DNBR analyses use methods approved by NRC (NMC topical report WPSRSEM P-A Revision 3).

The computer codes and methodologies used in each of the non-LOCA transient analyses are listed in Table RAI42-1 included below.

As indicated by Tables RAI42-2 through RAI42-6 and Tables RAI42-9 and RAI42-10, the NRC staff has approved all codes that were used in the non-LOCA transient analyses for Kewaunee. As for the applicable non-LOCA transient analysis methodologies, these have been reviewed and approved by the NRC staff via transient-specific topical reports (WCAPs) and/or through the review and approval of plant-specific safety analysis reports (see Table RAI42-1).

Code and methodology restrictions are specified in applicable SERs. Tables RAI42-2 through RAI42-6 and Tables RAI42-9 and RAI42-10 identify the SER conditions and restrictions for each of the computer codes listed in Table RAI42-1. Similarly, Tables RAI42-7 and RAI42-8 identify the SER conditions and restrictions for each methodology that has an approved topical report associated with it.

Tables RAI42-2 through RAI42-10 also provide the justifications for how each SER condition/restriction is satisfied in the Kewaunee analyses. To help ensure that all applicable SER conditions and restrictions are satisfied for each transient analysis that is performed, Westinghouse utilizes internal methodology guidance documents. Each analysis guidance document provides a description of the subject transient, a discussion of the plant protection systems that are expected to function, a list of the applicable event acceptance criteria, a list of the analysis input assumptions (e.g., directions of conservatism for initial condition values), a detailed description of the transient model development, and a discussion of the expected transient analysis results.

Although different codes and methods were applied in the new analyses, the application of these codes and methods to the KNPP licensing basis is valid (i.e., all restrictions and limitations of methodologies have been met).

| **Table RAI42-1:** | **Computer Codes and Methodologies Used in Non-LOCA Transient Analyses for Kewaunee** | | |
|---|---|---|---|
| **USAR Section** | **Event Description** | **Applicable Code(s)** | **Applicable Methodology** |
| 14.1.1 | Uncontrolled RCCA Withdrawal from a Subcritical Condition | TWINKLE (WCAP-7979-P-A), FACTRAN (WCAP-7908-A),<br>VIPRE (WCAP-14565-P-A) | SAR submittals |
| 14.1.2 | Uncontrolled RCCA Withdrawal at Power | RETRAN (WCAP-14882-P-A) | SAR submittals |
| 14.1.3 | RCCA Misalignment | LOFTRAN (WCAP-7907-P-A),<br>VIPRE (WCAP-14565-P-A)<br>ANC (WCAP-10965-P-A)<br>PHOENIX-P (WCAP-11596-P-A) | WCAP-11394-P-A |
| 14.1.4 | Chemical and Volume Control System Malfunction | N/A | SAR submittals |

| Table RAI42-1: | Computer Codes and Methodologies Used in Non-LOCA Transient Analyses for Kewaunee | | |
|---|---|---|---|
| **USAR Section** | **Event Description** | **Applicable Code(s)** | **Applicable Methodology** |
| 14.1.5 | Startup of an Inactive Reactor Coolant Loop | N/A | Event precluded by Tech Specs |
| 14.1.6 | Feedwater Temperature Reduction Incident | N/A | SAR submittals |
| 14.1.6 | Excessive Heat Removal Due to Feedwater System Malfunctions | RETRAN (WCAP-14882-P-A),<br>VIPRE (WCAP-14565-P-A) | SAR submittals |
| 14.1.7 | Excessive Load Increase Incident | N/A | SAR submittals |
| 14.1.8 | Loss of Reactor Coolant Flow | RETRAN (WCAP-14882-P-A),<br>VIPRE (WCAP-14565-P-A) | SAR submittals |
| 14.1.8 | Locked Rotor | RETRAN (WCAP-14882-P-A),<br>VIPRE (WCAP-14565-P-A),<br>FACTRAN (WCAP-7908-A) | SAR submittals |
| 14.1.9 | Loss of External Electrical Load | RETRAN (WCAP-14882-P-A) | SAR submittals |
| 14.1.10 | Loss of Normal Feedwater | RETRAN (WCAP-14882-P-A) | SAR submittals |
| 14.1.11 | Anticipated Transients Without Scram | N/A | N/A |
| 14.1.12 | Loss of AC Power to the Plant Auxiliaries | RETRAN (WCAP-14882-P-A) | SAR submittals |
| 14.2.5 | Steam Line Break | RETRAN (WCAP-14882-P-A),<br>VIPRE (WCAP-14565-P-A)<br>ANC (WCAP-10965-P-A)<br>PHOENIX-P (WCAP-11596-P-A) | SAR submittals |
| 14.2.6 | Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) | TWINKLE (WCAP-7979-P-A), FACTRAN (WCAP-7908-A) | WCAP-7588, Rev. 1-A |

| **Table RAI42-2:** Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - RETRAN | |
|---|---|
| Computer Code: | RETRAN |
| **Licensing Topical Report:** | WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999. |
| **Date of NRC Acceptance:** | February 11, 1999 (SER from F. Akstulewicz (NRC) to H. Sepp (Westinghouse)) |

**Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant**

**1.** ***"The transients and accidents that Westinghouse proposes to analyze with RETRAN are listed in this SER (Table 1) and the NRC staff review of RETRAN usage by Westinghouse was limited to this set. Use of the code for other analytical purposes will require additional justification."***

**Justification**

The transients listed in Table 1 of the SER are:

*Feedwater system malfunctions,*
*Excessive increase in steam flow,*
*Inadvertent opening of a steam generator relief or safety valve,*
*Steam line break,*
*Loss of external load/turbine trip,*
*Loss of offsite power,*
*Loss of normal feedwater flow,*
*Feedwater line rupture,*
*Loss of forced reactor coolant flow,*
*Locked reactor coolant pump rotor/sheared shaft,*
*Control rod cluster withdrawal at power,*
*Dropped control rod cluster/dropped control bank,*
*Inadvertent increase in coolant inventory,*
*Inadvertent opening of a pressurizer relief or safety valve,*
*Steam generator tube rupture.*

The transients analyzed for Kewaunee using RETRAN are:

*Uncontrolled RCCA withdrawal at power (USAR 14.1.1),*
*Excessive heat removal due to feedwater system malfunctions (USAR 14.1.6),*
*Loss of reactor coolant flow (USAR 14.1.8),*
*Locked rotor (USAR 14.1.8),*
*Loss of external electrical load (USAR 14.1.9),*
*Loss of normal feedwater (USAR 14.1.10),*
*Loss of AC power to the plant auxiliaries (USAR 14.1.12),*
*Steam line break (USAR 14.2.5).*

| **Table RAI42-2:** Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - RETRAN | |
|---|---|
| Computer Code: | RETRAN |
| **Licensing Topical Report:** | WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999. |
| **Date of NRC Acceptance:** | February 11, 1999 (SER from F. Akstulewicz (NRC) to H. Sepp (Westinghouse)) |
| **Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant** | |
| 2. | ***" WCAP-14882 describes modeling of Westinghouse designed 4-, 3, and 2-loop plants of the type that are currently operating. Use of the code to analyze other designs, including the Westinghouse AP600, will require additional justification."*** |
| | **Justification**<br>The Kewaunee Nuclear Power Plant is a 2-loop Westinghouse-designed plant that was "currently operating" at the time the SER was written (February 11, 1999). Therefore, additional justification is not required. |
| 3. | ***"Conservative safety analyses using RETRAN are dependent on the selection of conservative input. Acceptable methodology for developing plant-specific input is discussed in WCAP-14882 and in Reference 14 [WCAP-9272-P-A]. Licensing applications using RETRAN should include the source of and justification for the input data used in the analysis."*** |
| | **Justification**<br>The input data used in the RETRAN analyses performed by Westinghouse came from both NMC and Westinghouse sources. Assurance that the RETRAN input data is conservative for Kewaunee is provided via Westinghouse's use of transient-specific analysis guidance documents. Each analysis guidance document provides a description of the subject transient, a discussion of the plant protection systems that are expected to function, a list of the applicable event acceptance criteria, a list of the analysis input assumptions (e.g., directions of conservatism for initial condition values), a detailed description of the transient model development method, and a discussion of the expected transient analysis results. Based on the analysis guidance documents, conservative, plant-specific input values were requested and collected from the responsible NMC and Westinghouse sources. Consistent with the Westinghouse Reload Evaluation Methodology described in WCAP-9272-P-A, the safety analysis input values used in the Kewaunee analyses were selected to conservatively bound the values expected in subsequent operating cycles. |

| **Table RAI42-3:** Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - TWINKLE | |
|---|---|
| Computer Code: | TWINKLE |
| **Licensing Topical Report:** | WCAP-7979-P-A, "TWINKLE – A Multidimensional Neutron Kinetics Computer Code," January 1975. |
| **Date of NRC Acceptance:** | July 29, 1974 (SER from D. B. Vassallo (U.S. Atomic Energy Commission) to R. Salvatori (Westinghouse)) |
| **Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant** | |
| | ***There are no conditions, restrictions, or limitations cited in the TWINKLE SER.*** |
| | **Justification**<br>As the TWINKLE SER does not cite any conditions, restrictions, or limitations, additional justification is not required. |

| **Table RAI42-4:** Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - FACTRAN | |
|---|---|
| Computer Code: | FACTRAN |
| **Licensing Topical Report:** | WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a $UO_2$ Fuel Rod," December 1989. |
| **Date of NRC Acceptance:** | September 30, 1986 (SER from C. E. Rossi (NRC) to E. P. Rahe (Westinghouse)) |
| **Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant** | |
| **1.** | ***"The fuel volume-averaged temperature or surface temperature can be chosen at a desired value which includes conservatisms reviewed and approved by the NRC."*** |
| | **Justification**<br>The FACTRAN code was used in the analyses of the following transients for Kewaunee: Uncontrolled RCCA Withdrawal from a Subcritical Condition (USAR 14.1.1), Locked Rotor (USAR 14.1.8), and RCCA Ejection (USAR 14.2.6). Initial fuel temperatures were used as FACTRAN input in the Locked Rotor and RCCA Ejection analyses. The assumed fuel temperatures for these transients were calculated using the NRC-approved PAD 4.0 computer code (see WCAP-15063-P-A). As indicated in WCAP-15063-P-A, the NRC has approved the method of determining uncertainties for PAD 4.0 fuel temperatures. |
| **2.** | ***"Table 2 presents the guidelines used to select initial temperatures."*** |
| | **Justification**<br>In summary, Table 2 of the SER specifies that the initial fuel temperatures assumed in the FACTRAN analyses of the following transients should be "High" and include uncertainties: Loss of Flow, Locked Rotor, and Rod Ejection. As discussed above, fuel temperatures were used as input to the FACTRAN code in the Locked Rotor and RCCA Ejection analyses for Kewaunee. The assumed fuel temperatures, which were calculated using the PAD 4.0 computer code (see WCAP-15063-P-A), include uncertainties and are conservatively high. |

| **Table RAI42-4:** Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - FACTRAN | |
|---|---|
| Computer Code: | FACTRAN |
| **Licensing Topical Report:** | WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a $UO_2$ Fuel Rod," December 1989. |
| **Date of NRC Acceptance:** | September 30, 1986 (SER from C. E. Rossi (NRC) to E. P. Rahe (Westinghouse)) |
| **Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant** | |
| **3.** | ***"The gap heat transfer coefficient may be held at the initial constant value or can be varied as a function of time as specified in the input."*** |
| | **Justification**<br>The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2. For the RCCA Withdrawal from a Subcritical Condition transient, the gap heat transfer coefficient is kept at a conservative constant value throughout the transient; a high constant value is assumed to maximize the peak heat flux (for DNB concerns) and a low constant value is assumed to maximize fuel temperatures. For the Locked Rotor and RCCA Ejection transients, the initial gap heat transfer coefficient is based on the predicted initial fuel surface temperature, and is ramped rapidly to a very high value at the beginning of the transient to simulate clad collapse onto the fuel pellet. |
| **4.** | ***"...the Bishop-Sandberg-Tong correlation is sufficiently conservative and can be used in the FACTRAN code. It should be cautioned that since these correlations are applicable for local conditions only, it is necessary to use input to the FACTRAN code which reflects the local conditions. If the input values reflecting average conditions are used, there must be sufficient conservatism in the input values to make the overall method conservative."*** |
| | **Justification**<br>Local conditions related to temperature, heat flux, peaking factors and channel information were input to FACTRAN for each transient analyzed for Kewaunee (RCCA Withdrawal from a Subcritical Condition (USAR 14.1.1), Locked Rotor (USAR 14.1.8), RCCA Ejection (USAR 14.2.6)). Therefore, additional justification is not required. |
| **5.** | ***"The fuel rod is divided into a number of concentric rings. The maximum number of rings used to represent the fuel is 10. Based on our audit calculations we require that the minimum of 6 should be used in the analyses."*** |
| | **Justification**<br>At least 6 concentric rings were assumed in FACTRAN for each transient analyzed for Kewaunee (RCCA Withdrawal from a Subcritical Condition (USAR 14.1.1), Locked Rotor (USAR 14.1.8), RCCA Ejection (USAR 14.2.6)). |

<table>
<tr><th colspan="2">Table RAI42-4: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - FACTRAN</th></tr>
<tr><td>Computer Code:</td><td>FACTRAN</td></tr>
<tr><td>Licensing Topical Report:</td><td>WCAP-7908-A, “FACTRAN – A FORTRAN IV Code for Thermal Transients in a $UO_2$ Fuel Rod,” December 1989.</td></tr>
<tr><td>Date of NRC Acceptance:</td><td>September 30, 1986 (SER from C. E. Rossi (NRC) to E. P. Rahe (Westinghouse))</td></tr>
<tr><th colspan="2">Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant</th></tr>
<tr><td>6.</td><td>“Although time-<u>independent</u> mechanical behavior (e.g., thermal expansion, elastic deformation) of the cladding are considered in FACTRAN, time-<u>dependent</u> mechanical behavior (e.g., plastic deformation) is not considered in the code. ...for those events in which the FACTRAN code is applied (see Table 1), significant time-dependent deformation of the cladding is not expected to occur due to the short duration of these events or low cladding temperatures involved (where DNBR Limits apply), or the gap heat transfer coefficient is adjusted to a high value to simulate clad collapse onto the fuel pellet.”</td></tr>
<tr><td></td><td><u>Justification</u><br>The three transients that were analyzed with FACTRAN for Kewaunee (RCCA Withdrawal from a Subcritical Condition (USAR 14.1.1), Locked Rotor (USAR 14.1.8), and RCCA Ejection (14.2.6)) are included in the list of transients provided in Table 1 of the SER; each of these transients is of short duration. For the RCCA Withdrawal from a Subcritical Condition transient, relatively low cladding temperatures are involved, and the gap heat transfer coefficient is kept constant throughout the transient. For the Locked Rotor and RCCA Ejection transients, a high gap heat transfer coefficient is applied to simulate clad collapse onto the fuel pellet. The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2.</td></tr>
<tr><td>7.</td><td>“The one group diffusion theory model in the FACTRAN code slightly overestimates at beginning of life (BOL) and underestimates at end of life (EOL) the magnitude of flux depression in the fuel when compared to the LASER code predictions for the same fuel enrichment. The LASER code uses transport theory. There is a difference of about 3 percent in the flux depression calculated using these two codes. When [T(centerline) – T(Surface)] is on the order of 3000°F, which can occur at the hot spot, the difference between the two codes will give an error of 100°F. When the fuel surface temperature is fixed, this will result in a 100°F lower prediction of the centerline temperature in FACTRAN. We have indicated this apparent nonconservatism to Westinghouse. In the letter NS-TMA-2026, dated January 12, 1979, Westinghouse proposed to incorporate the LASER-calculated power distribution shapes in FACTRAN to eliminate this non-conservatism. We find the use of the LASER-calculated power distribution in the FACTRAN code acceptable.”</td></tr>
<tr><td></td><td><u>Justification</u><br>The condition of concern (T(centerline) – T(surface) on the order of 3000°F) is expected for transients that reach, or come close to, the fuel melt temperature. As this applies only to the RCCA ejection transient, the LASER-calculated power distributions were used in the FACTRAN analysis of the RCCA ejection transient for Kewaunee.</td></tr>
</table>

<table>
<tr><td colspan="2">Table RAI42-5: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - LOFTRAN</td></tr>
<tr><td>Computer Code:</td><td>LOFTRAN</td></tr>
<tr><td>Licensing Topical Report:</td><td>WCAP-7907-P-A, "LOFTRAN Code Description," April 1984.</td></tr>
<tr><td>Date of NRC Acceptance:</td><td>July 29, 1983 (SER from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse))</td></tr>
<tr><td colspan="2">Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant</td></tr>
<tr><td>1.</td><td>"LOFTRAN is used to simulate plant response to many of the postulated events reported in Chapter 15 of PSARs and FSARs, to simulate anticipated transients without scram, for equipment sizing studies, and to define mass/energy releases for containment pressure analysis. The Chapter 15 events analyzed with LOFTRAN are:<br><br>
- Feedwater System Malfunction<br>
- Excessive Increase in Steam Flow<br>
- Inadvertent Opening of a Steam Generator Relief or Safety Valve<br>
- Steamline Break<br>
- Loss of External Load<br>
- Loss of Offsite Power<br>
- Loss of Normal Feedwater<br>
- Feedwater Line Rupture<br>
- Loss of Forced Reactor Coolant Flow<br>
- Locked Pump Rotor<br>
- Rod Withdrawal at Power<br>
- Rod Drop<br>
- Startup of an Inactive Pump<br>
- Inadvertent ECCS Actuation<br>
- Inadvertent Opening of a Pressurizer Relief or Safety Valve<br><br>
This review is limited to the use of LOFTRAN for the licensee safety analyses of the Chapter 15 events listed above, and for a steam generator tube rupture..."</td></tr>
<tr><td></td><td><u>Justification</u><br>The LOFTRAN code was only used in the analysis of the Rod Drop transient (USAR 14.1.3) for Kewaunee. As this transient matches one of the transients listed in the SER, additional justification is not required.</td></tr>
</table>

| Table RAI42-6: Approval Status & SER Requirements for Core Analysis Codes - VIPRE | |
|---|---|
| Computer Code: | VIPRE |
| **Licensing Topical Report:** | WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis" October 1999. |
| **Date of NRC Acceptance:** | January 19, 1999 (SER from T. H. Essig (NRC) to H. Sepp (Westinghouse)) |
| **Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant** | |
| 1. | ***"Selection of the appropriate CHF correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal."*** |
| | **Justification**<br>The WRB-1 correlation with a 95/95 correlation limit of 1.17 was used in the DNB analyses for the Kewaunee 422V+ fuel. The validity of the WRB-1 DNB correlation applicability to the 422V+ fuel was discussed in the response to Attachment 3 RAI No. 21.<br>The use of the plant specific hot channel factors and other fuel dependent parameters in the DNB analysis for the Kewaunee 422V+ fuel are the same as those previously used and approved for the safety evaluations of other Westinghouse two-loop plants using the same fuel design. |
| 2. | ***"Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE."*** |
| | **Justification**<br>The core boundary conditions for the VIPRE calculations for the 422V+ fuel are all generated from NRC-approved codes and analysis methodologies. Conservative reactor core boundary conditions were justified for use as input to VIPRE as discussed in the RTSR. Continued applicability of the input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology WCAP-9272-P-A. |
| 3. | ***"The NRC Staff's generic SER for VIPRE set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification."*** |
| | **Justification**<br>As discussed in response to Condition 1, the WRB-1 correlation with a limit of 1.17 was used for the DNB analyses of 422V+ fuel in Kewaunee. For conditions where WRB-1 is not applicable, the W-3 DNB correlation was used with a limit of 1.30 (1.45 if $500 < P < 1000$). |

| **Table RAI42-6:** Approval Status & SER Requirements for Core Analysis Codes - VIPRE | |
|---|---|
| Computer Code: | VIPRE |
| **Licensing Topical Report:** | WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis" October 1999. |
| **Date of NRC Acceptance:** | January 19, 1999 (SER from T. H. Essig (NRC) to H. Sepp (Westinghouse)) |
| **Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant** | |
| 4. | ***"Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC Staff's generic review of VIPRE did not extent to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC Staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained."*** |
| | **Justification**<br>The application of the VIPRE to the 422V+ fuel upgrade in Kewaunee did not include usage in the post-CHF region. |

| **Table RAI42-7:** Approval Status & SER Requirements for Non-LOCA Transient Analysis Methods – Dropped Rod | |
|---|---|
| Transient: | RCCA Misalignment (Dropped Rod) |
| **Licensing Topical Report:** | WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," January 1990. |
| **Date of NRC Acceptance:** | October 23, 1989 (SER from A. C. Thadani (NRC) to R. A. Newton (WOG)) |
| **Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant** | |
| 1. | ***"The Westinghouse analysis, results and comparisons are reactor and cycle specific. No credit is taken for any direct reactor trip due to dropped RCCA(s). Also, the analysis assumes no automatic power reduction features are actuated by the dropped RCCA(s). A further review by the staff (for each cycle) is not necessary, given the utility assertion that the analysis described by Westinghouse has been performed and the required comparisons have been made with favorable results."*** |
| | **Justification**<br>For the reference cycle assumed in the Kewaunee 422V+ fuel transition/uprate program, it is affirmed that the methodology described in WCAP-11394-P-A was performed and the required comparisons have been made with acceptable results (DNB limits are not exceeded). |

| **Table RAI42-8:** Approval Status & SER Requirements for Non-LOCA Transient Analysis Methods – RCCA Ejection | |
|---|---|
| Transient: | RCCA Ejection |
| **Licensing Topical Report:** | WCAP-7588 Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," January 1975. |
| **Date of NRC Acceptance:** | August 28, 1973 (SER from D. B. Vassallo (AEC) to R. Salvatori (Westinghouse)) |
| **Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant** | |
| **1.** | ***"The staff position, as well as that of the reactor vendors over the last several years, has been to limit the average fuel pellet enthalpy at the hot spot following a rod ejection accident to 280 cal/gm. This was based primarily on the results of the SPERT tests which showed that, in general, fuel failure consequences for $UO_2$ have been insignificant below 300 cal/gm for both irradiated and unirradiated fuel rods as far as rapid fragmentation and dispersal of fuel and cladding into the coolant are concerned. In this report, Westinghouse has decreased their limiting fuel failure criterion from 280 cal/gm (somewhat less than the threshold of significant conversion of the fuel thermal energy to mechanical energy) to 225 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods. Since this is a conservative revision on the side of safety, the staff concludes that it is an acceptable fuel failure criterion."*** |
| | **Justification**<br>**The maximum fuel pellet enthalpy at the hot spot calculated for each Kewaunee-specific RCCA Ejection case is less than 200 cal/gm. These results satisfy the fuel failure criterion accepted by the staff.** |
| **2.** | ***"Westinghouse proposes a clad temperature limitation of 2700°F as the temperature above which clad embrittlement may be expected. Although this is several hundred degrees above the maximum clad temperature limitation imposed in the AEC ECCS Interim Acceptance Criteria, this is felt to be adequate in view of the relatively short time at temperature and the highly localized effect of a reactivity transient."*** |
| | **Justification**<br>As discussed in Westinghouse letter NS-NRC-89-3466 written to the NRC (W. J. Johnson to R. C. Jones, dated October 23, 1989), the 2700°F clad temperature limit was historically applied by Westinghouse to demonstrate that the core remains in a coolable geometry during an RCCA ejection transient. This limit was never used to demonstrate compliance with fuel failure limits and is no longer used to demonstrate core coolability. The RCCA ejection acceptance criteria applied by Westinghouse to demonstrate long term core coolability and compliance with applicable offsite dose requirements are those defined in the suggested revisions to KNPP USAR Section 14.2.6 (fuel pellet enthalpy, RCS pressure, and fuel melt). |

| **Table RAI42-9:** Approval Status & SER Requirements for Core Analysis Codes - ANC | |
|---|---|
| Computer Code: | ANC |
| **Licensing Topical Report:** | WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code" September 1986. |
| **Date of NRC Acceptance:** | June 23, 1986 (SER from C. Berlinger (NRC) to E. P. Rahe (Westinghouse)) |
| **Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant** | |
| | ***Although there are no conditions, restrictions, or limitations explicitly cited in the ANC SER, the SER does conclude that "the ANC code provides an accurate calculation of core reactivity, reactivity coefficients critical boron, rod worths and core power distribution for use in design and safety analyses."*** |
| | **Justification**<br>In support of the Kewaunee fuel transition, the ANC code was used to calculate power distributions for normal (design) and off-normal (safety analysis) conditions, and was also used for reactivity calculations. As these code applications are consistent with those listed in the SER, additional justification is not required. |

| **Table RAI42-10:** Approval Status & SER Requirements for Core Analysis Codes – PHOENIX-P | |
|---|---|
| Computer Code: | PHOENIX-P |
| **Licensing Topical Report:** | WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores" June 1988. |
| **Date of NRC Acceptance:** | May 17, 1988 (SER from A. C. Thadani (NRC) to R. W. Johanson (Westinghouse)) |
| **Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant** | |
| | ***Although there are no conditions, restrictions, or limitations explicitly cited in the PHOENIX-P/ANC SER, the SER does conclude that "The PHOENIX-P lattice physics methods and the PHOENIX-P/ANC nodal analysis systems described in this report are thus acceptable for use in PWR design analysis."*** |
| | **Justification**<br>As Kewaunee is a PWR plant, it is acceptable to use the PHOENIX-P and PHOENIX-P/ANC code system, and additional justification is not required. |

**43. Provide a tabulation of the thermal design parameters compare to values assumed in safety analyses to demonstrate that proper conservativeness are available for the safety analyses assumptions.**

**NMC Response:**

The table below demonstrates that proper conservativeness is available for the safety analysis input assumptions. The safety analysis parameter values bound the actual plant (expected best estimate values) at the uprated power conditions.

| Parameter | 1772 MWt (BE 7.4% Uprate) | Uprate Safety Analysis (includes uncertainties) |
|---|---|---|
| Core Power (MWt) | 1772 | 1783 |
| NSSS Power (MWt) | 1779.1 | 1790 |
| RCS Flow Rate (gpm/loop) | 99,000 | 89,000 |
| RCS Pressure (psia) | 2250 | 2200 |
| Vessel Inlet Temp (°F) | 541.4 | 545.2 |
| Primary Avg. Temp (Tavg-°F) | 572 | 579.0 |
| Vessel Outlet Temp (°F) | 602.6 | 612.8 |

The stretch power uprate license amendment request supports a rated core power of 1772 MWt with a power uncertainty of 0.6%, even though some of the non-LOCA analyses were conservatively performed assuming up to a 2% power uncertainty.

**44. Describe changes of NSSS design transients for KNPP operating at 1772 MWt.**

**NMC Response:**

The only formal design transient revisions made were in the feedwater temperature responses. For the remainder of the design transient parameters (T-hot, T-cold, RCS flow and pressure, pressurizer pressure, steam generator steam temperature, steam and feedwater flows), the restriction was made that the full power steam pressure was no lower than the 644 psia steam pressure. With this restriction the current KNPP NSSS design transients remained as they currently exist.

The reporting format of the design transients was changed as being a "change from initial" format for T-hot, T-cold and the steam generator steam temperature from the previous format of reporting absolute temperatures. This is consistent with the format of reporting these parameters used in more recent plant designs. This format more readily

allows for the accommodation of plant changes if needed. This resulted in a re-plotting of the existing transients, however no technical changes were made.

## Questions from Probabilistic Safety Assessment Branch - Heating, Ventilation, and Air Conditioning (HVAC)

45. **Provide additional information regarding the potential impact of the stretch power uprate (SPU) on those HVAC systems discussed in the Standard Review Plan sections 6.4, 6.5.1, 9.4.1, 9.4.2 and 9.4.5. This should include a discussion of the impact, if any, during both normal and post-accident operations resulting from increases in heat loads due to SPU and the bases for your determination of system acceptability post-SPU.**

**NMC Response**

Kewaunee (KNPP) was designed and licensed before the NRC's Standard Review Plan (SRP) was developed. Therefore, the KNPP Technical Specifications and system designations do not consistently match the HVAC system designations referenced in the SRP.

At KNPP, the Auxiliary Building Ventilation System (reference Attachment 4 of our submittal, page 8-53) has separate ventilation systems for equipment rooms, spent fuel pool, containment penetrations, control room, and non-radioactive areas. This one system crosses all the SRP section boundaries referenced in the NRC question.

In regards to SRP sections 6.4 and 9.4.1, dealing with the control room habitability and ventilation system, the impact of the power uprate, including accident scenarios, can be found in the following sections of Attachment 4 of our submittal: page 6-185 and Table 6.7-3 (page 6-219), the "Results and Conclusion" sections for each accident analyzed (6.7.2.3, 6.7.3.3, 6.7.4.3, 6.7.5.3, 6.7.6.3, 6.7.7.3, 6.7.8.3, and 6.7.9.3) and Section 8.3.13 (pages 8-53, 8-57, 8-60, 8-61).

In regards to SRP sections 6.5.1 and 9.4.5, dealing with the ESF Atmosphere Cleanup and ESF Ventilation Systems, the impact of the power uprate, including accident scenarios, can be found in the following sections of Attachment 4 of our submittal: 6.7.6.1.3 (Activity Removal from the Containment Atmosphere) and various sub-sections of 8.3.13 (pages 8-54, 8-55, 8-56, 8-58, 8-60, 8-61) for Auxiliary Building Special Ventilation System, Containment Vessel Air Handling System, and Shield Building Ventilation System.

In regards to SRP sections 9.4.2, dealing with the Spent Fuel Pool ventilation, the impact of the power uprate, including accident scenarios, can be found in the following section of Attachment 4 of our submittal: 8.3.13 (pages 8-54, 8-57, 8-60).

Radiological consequences of the stretch power uprate to the various plant areas are discussed in Section 8.8.1.3 (pages 8-126 through 8-128).

**Questions from Plant Systems - BOP**

46. **Regulatory Application: Please provide a discussion of the regulatory bases that are applicable to the power uprate request for the ultimate heat sink.**

**NMC Response:**

The U.S. Atomic Energy Commission (AEC) issued a "Safety Evaluation of the Kewaunee Nuclear Power Plant" on July 24, 1972, supplemented December 18, 1972, and May 10, 1973. The AEC performed a technical review of the KNPP against the General Design Criteria (GDC) in effect at the time (those published in February of 1971 and revised in July of 1971) and concluded that the KNPP design generally conforms to the intent of these GDC . How KNPP satisfies these GDC's is stated in section 1.8 of the KNPP Updated Safety Analysis Report (USAR) and in the applicable sections.

The primary regulatory bases applicable to the Ultimate Heat Sink (UHS) (for KNPP the UHS is Lake Michigan) are 10 CFR 50.34, Contents of Applications: Technical Information and 10 CFR 50 Appendix A, General Design Criteria (GDC) for Nuclear Power Plants as modified by the 1972 KNPP Safety Evaluation. The applicable GDC, as described in NUREG 0800, Standard Review Plan (SRP) section 9.2.5, "Ultimate Heat Sink," are GDC 2, "Design Bases for Protection Against Natural Phenomena," GDC 5, "Sharing of Structures, Systems, and Components," GDC 44, "Cooling Water," GDC 45, "Inspection of Cooling Water System," and GDC 46, "Testing of Cooling Water System."

In the KNPP USAR for GDC 2 (USAR Section 1.8) it states, in part, that the systems and components designated Class I in Appendix B are designed to withstand, without loss of capability to protect the public, the most severe environmental phenomena ever experienced at the site with appropriate margins included in the design for uncertainties in historical data. In USAR Table B.2-1 it states that the intake piping to the screenhouse, the screenhouse (Areas Housing Service Water Facilities, Equipment, and Piping), and the Service Water System serving Class I equipment is Class I.

In the KNPP USAR for GDC 5 (USAR Section 1.8 GDC 4) it states that analyses confirmed that the sharing of components among systems does not result in interference with the basic function and operability of these systems and, hence, there is no undue risk to the health and safety of the public. Those systems or components, which are shared functionally within the plant, are designed in such a manner that the sharing does not impair plant safety.

In the KNPP USAR GDC's 44, 45, 46, and 47 (USAR Section 1.8) describes KNPP's applicability to draft GDC's that were noticed in July of 1967. These GDC's were later modified to those contained in 10 CFR 50 Appendix A in February of 1971 as GDC's 44, 45, and 46. Differences between the 1967 draft GDC's and the 1971 GDC's include changing "Emergency Core Cooling System" to "Cooling System," requiring two ECCS to a system with suitable redundancy for a single failure, and testing of systems and components was combined into one criteria. The KNPP USAR states that by combining the use of passive accumulators with two independent high pressure pumping systems and two independent low-pressure pumping systems, abundant emergency core cooling

is provided even if there should be a failure of any component in any system. Also, the USAR states that critical portions are accessible for examination by visual, optical or other nondestructive means, the emergency core cooling system design permits periodic testing of active components for operability and required functional performance, and by recirculation to the refueling water storage tank, the Emergency Core Cooling System delivery capability can be tested periodically and can be tested to the last valve before the piping enters the reactor coolant piping.

Lake Michigan, as the source of water for the SW System, is the UHS. The SW System either directly cools emergency core cooling equipment, or provides the cooling water to the Component Cooling System (CC) heat exchangers, and the CC System removes heat from the core cooling systems. As discussed in questions #47a and #47b below, the acceptance criteria for the design basis accidents and transients for which the UHS is important to accident mitigation are shown to be satisfied at the SUR conditions. Therefore the UHS for KNPP is acceptable as shown by the licensing basis safety analyses.

**47. Ultimate Heat Sink**

**(a) From the standpoint of the proposed power uprate, provide a full description and details of the ultimate heat sink capability for KNPP.**

**NMC Response:**

The Service Water (SW) System supplies water from Lake Michigan (the Ultimate Heat Sink) for cooling equipment in the steam plant, Containment fan coil units, and Reactor auxiliary systems. The purpose of the SW System is to provide redundant cooling water supplies for the engineered safeguards equipment required during post accident conditions and for single non redundant supplies to other systems, including balance of plant equipment. The SW System utilizes:

Four pumps with two redundant headers,
Strainers,
Isolation valves, and
Four Traveling Water Screens.

These components are powered from emergency buses, with the exception of Traveling Water Screens 1A2 and 1B1, which are powered from non-emergency buses.

The SW System is designed with two redundant headers, each capable of providing for all anticipated post accident heat removal requirements, including leakage allowances, at the Ultimate Heat Sink (UHS) temperature limit of 80°F. Each header is capable of supplying cooling water to meet balance of plant (BOP) equipment requirements. The SW System is also designed to automatically start the necessary number of pumps to maintain adequate system pressure.

Various, redundant safeguard equipment and coolers are supplied with SW from each Aux Bldg header. Examples of major equipment include: four Containment fan coil units, the Auxiliary Feedwater (AFW) Pumps, and the Component Cooling (CC) water heat exchangers. Some non-safety Class 1 and non-redundant coolers and equipment are supplied from SW headers. Some examples include the CW Pump seals and bearing coolers, Traveling Water Screen wash nozzles, Fire Protection Jockey Pump, and Spent Fuel Pool Emergency Makeup Valve. The Main Service Water Header supplies the Diesel Generator Cooler Heat Exchanger.

As discussed in Section 8.3.10, "Service Water," of WCAP-16040-NP, "NSSS/BOP Licensing Report," the required SW flow rates to engineered safeguards equipment for accident conditions are not impacted by the 7.4% uprate, since the current analysis was based on conditions that are still bounding. The most significant impact of the uprate is the increase of the Turbine Building flow requirements for normal full power conditions. The combination of the uprate and the increased SW maximum temperature of 80°F increased the required flow to the Turbine Building header by about 73% above current requirements with 66°F SW for normal full power operation. However, SW System requirements still continue to be met. The other SW heat loads, such as CCW HXs, EDGs, SFP HX, and area FCUs, do not require any increase in SW flow for normal and accident conditions above those already established for the current power level.

(b) **Confirm that the existing design-basis ultimate heat sink temperature limit remains valid based on post licensing data trends (e.g., air and water temperatures, wind speed, water volume).**

**NMC Response:**

The Ultimate Heat Sink (UHS) is Lake Michigan, and the design basis UHS temperature limit is 80°F. The Service Water System (SWS) is the KNPP system that ultimately cools the reactor and the containment in the event of a design basis accident. The maximum (SWS) temperature recorded at the plant was 76°F during the early 1990's.

To support the stretch power uprate (SUR) design basis containment post accident long term (30 day) cooling analysis (containment long term response analysis is used for equipment qualification), an additional calculation was performed which examined the maximum average lake temperatures occurring during the months of June through September for the last twelve years. This calculation, using the methodology described in Reg Guide 1.27, showed the following maximum average lake water temperatures:

| | |
|---|---|
| 1-hour | 74.70°F |
| 1-day | 72.94°F |
| 5-day | 72.83°F |
| 30-day | 69.55°F |

The containment long term cooling analysis uses the resulting temperatures from the average lake temperature calculation. All other SSC's continue to use the design basis SWS temperature limit of 80°F. Based on analysis of historical lake temperature data, this design basis UHS temperature limit remains acceptable.

48. **In reference to the spent fuel pool cooling system (Attachment 4, Section 8.3.8), please explain how you meet the following for the power uprate:**

a. **Demonstrate adequate SFP cooling capacity by either performing a bounding evaluation or committing to a method of performing outage-specific evaluations.**

**NMC Response:**

To ensure adequate SFP cooling capacity following stretch power uprate, a bounding analysis was initially performed which showed that the SFP temperature would increase to approximately 163°F, which is above the design temperature of 150°F. Therefore, KNPP has chosen to perform a cycle specific heat load calculation prior to each refueling outage. This calculation determines the minimum in-core hold time (ICHT) required to ensure maintaining the SFP temperature below 150°F.

b. **If a bounding calculation was performed, demonstrate adequate SFP cooling capacity for two scenarios: (1) full cooling capability and (2) a single-failure of an active cooling system component.**

**NMC Response:**

A bounding calculation was performed, however, the analysis showed that the SFP temperature would increase to approximately 163°F, which is above the design temperature of 150°F. Therefore, KNPP has chosen to perform a cycle specific heat load calculation prior to each refueling outage, and this question is not applicable.

c. **For full cooling capability evaluation, demonstrate that the following analysis conditions are met: (1) decay heat load is calculated based on bounding estimates of offload size, decay time, power history, and inventory of previously discharged assemblies; (2) heat removal capability is based on bounding estimates of ultimate heat sink temperature, cooling system flow rates, and heat exchanger performance (e.g., fouling and tube plugging margin); (3) alternate heat removal paths (e.g., evaporative cooling) must be appropriately validated and based on bounding input parameter values (e.g., air temperature, relative humidity, and ventilation flow rate); (4) actual bulk SFP temperature must remain below 140 °F - calculated SFP temperatures up to approximately 150 °F are acceptable when justified by conservative methods or assumptions; and (5) with appropriate administrative controls to verify that analysis inputs bound actual conditions, a set of bounding analyses may be prepared to support operational flexibility.**

NMC Response:

This question also is not applicable because KNPP has chosen to perform a cycle specific heat load calculation prior to each refueling outage.

d. **For single active failure evaluation, demonstrate that the following analysis conditions are met: (1) decay heat load is calculated based on a bounding estimate of offload size, decay time, power history, and inventory of previously discharged assemblies; (2) heat removal capability is based on a bounding estimate of ultimate heat sink temperature, heat exchanger performance (e.g., fouling and tube plugging margin), and cooling system flow rates assuming the limiting single-failure with regard to heat removal capability; (3) alternate heat removal paths (e.g., evaporative cooling) must be appropriately validated and based on bounding input parameter values (e.g., air temperature, relative humidity, and ventilation flow rate); (4) calculated bulk SFP temperature must remain below the design temperature of the SFP structure and liner, and calculated peak storage cell temperature must remain below the storage rack design temperature; (5) for plants where a single-failure results in a complete loss of forced cooling, the analysis should demonstrate that the loss of cooling would be identified and forced cooling would be restored before the bounding decay heat load would cause the SFP temperature to reach its design limit; and (6) with appropriate administrative controls to verify that analysis inputs bound actual conditions, a set of bounding analyses may be prepared to support operational flexibility.**

NMC Response:

This question also is not applicable because KNPP has chosen to perform a cycle specific heat load calculation prior to each refueling outage.

e. **If you choose to define a method to calculate operational limits prior to every offload using the anticipated actual conditions at the time of the offload, demonstrate that the following cycle-specific conditions are met: (1) define the method to calculate decay heat load based on decay time, power history, and inventory of previous fuel discharges; (2) define the method to calculate cooling system heat removal capacity based on ultimate heat sink temperature, cooling system flow rates, and heat exchanger performance parameters; (3) define the method for calculating alternate heat removal capability (e.g., evaporative cooling) and provide validation of the method; (4) using the methods defined to calculate heat load and heat removal capability, define the method to determine the limiting value of the variable operational parameter (typically, decay time) such that bulk SFP temperature will remain below 140 °F with full cooling capability; (5) using the methods defined to calculate heat load and heat removal capability, define the method to determine the limiting value of the variable operational parameter (typically, decay time) such that bulk SFP temperature will be maintained below the SFP structure design temperature assuming a singlefailure affecting the forced cooling system (this may be a heat balance analysis if cooling is degraded or a heatup rate analysis if**

**forced cooling is completely lost and subsequently recovered using redundant components); and (6) describe administrative controls that will be implemented each offload to ensure the cyclespecific analysis inputs and results bound actual conditions prior to fuel movement.**

**NMC Response:**

**(1) define the method to calculate decay heat load based on decay time, power history, and inventory of previous fuel discharges;**

The method to determine decay heat was based on a previous analysis by Holtec International (HI-992245, "Bulk Temperature Analysis for the Kewaunee Spent Fuel Pools and Transfer Canal") using Holtec's QA validated LONGOR program, which incorporated the ORIGEN2 computer code for decay heat calculations. This methodology was previously reviewed and approved by the NRC in support of Technical Specification Amendment 150. This previous analysis was used for the uprate analysis with the following modifications.

- Fuel assemblies with discharge dates of Spring 2000, Fall 2001, and Spring 2003 (pre-uprate) had their decay heat reduced by a factor of 4.3%. The HOLTEC report assumed a power uprate of 4.3% for spent fuel discharged after 1999, and the uprate analysis used the actual value of 6.0% for spent fuel discharged after 2003. The 1.4% portion of the uprate was accounted for in the HOLTEC report.
- The Spring 2003 refueling was assumed to consist of 48 fuel assemblies.
- The assemblies from the Spring 2003 and Fall 2004 were assumed to have a decay heat consistent with an assembly average discharge burnup of 60,000 MWD/MTU.
- Fuel assemblies discharged after 2002 would come from a 1772 MW core. This value would be increased by 0.6% uncertainty (reduction in uncertainty due to Appendix K portion of the uprate) for an analysis value of 1782.63 MW.
- Fuel assemblies with discharge dates after 2002 would have their decay heat increased by an approximate factor of 1.55% (1782.63 / 1755.42) to reflect the uprate conditions (including the updated 0.6% calorimetric uncertainty).

**(2) define the method to calculate cooling system heat removal capacity based on ultimate heat sink temperature, cooling system flow rates, and heat exchanger performance parameters;**

Cooling system heat removal capacity was calculated using Proto-HX, a QA-1 validated computer program for analyzing heat exchanger performance based on the manufacturer's specification sheet. Conservative values for flow rates, temperatures, and performance parameters were used to determine heat transfer rates.

**(3) define the method for calculating alternate heat removal capability (e.g., evaporative cooling) and provide validation of the method;**

The evaporative cooling calculation was performed by Holtec International (HI-992245, "Bulk Temperature Analysis for the Kewaunee Spent Fuel Pools and Transfer Canal"). The calculation is proprietary, however, this methodology was previously reviewed and approved by the NRC in support of Technical Specification Amendment 150.

**(4) using the methods defined to calculate heat load and heat removal capability, define the method to determine the limiting value of the variable operational parameter (typically, decay time) such that bulk SFP temperature will remain below 140 °F with full cooling capability;**

The minimum in-core hold time (ICHT) is calculated such that the heat removal rate meets or exceeds the decay heat input resulting in a spent fuel pool temperature less than the design temperature of 150°F.

**(5) using the methods defined to calculate heat load and heat removal capability, define the method to determine the limiting value of the variable operational parameter (typically, decay time) such that bulk SFP temperature will be maintained below the SFP structure design temperature assuming a single failure affecting the forced cooling system (this may be a heat balance analysis if cooling is degraded or a heatup rate analysis if forced cooling is completely lost and subsequently recovered using redundant components);**

The same methodology as stated above is used to determine ICHT. For example, on the loss of one spent fuel pool pump, a reduced spent fuel pool flow rate would be used in the heat removal rate analysis.

**(6) describe administrative controls that will be implemented each offload to ensure the cycle-specific analysis inputs and results bound actual conditions prior to fuel movement.**

KNPP Refueling Procedure RF-01.00 requires determination of spent fuel pool heat exchanger performance and resulting minimum ICHT prior to initiating core offload.

**f. Following a loss-of-SFP cooling event, demonstrate the ability to provide two sources of make-up water prior to the occurrence of boiling in the pool. Assuming the worst single-active failure occurred, demonstrate the licensee has a process to determine the time to boil assuming the initial pool temperature is the peak temperature from a planned offload.**

**NMC Response:**

In the event of a loss of the Spent Fuel Pool Cooling System, the time for the spent fuel pool temperature to rise from the maximum allowed temperature of 150°F (therefore the peak temperature from a planned offload) to boiling at 212°F would be approximately 6.5 hours. This is sufficient time to establish normal makeup from the Chemical and Volume Control System, or emergency makeup

from Service Water. Steps to establish both makeup paths are in current operating procedures.

g. **Demonstrate that at least one make-up source has a capacity that is equal to or greater than the calculated boil-off rate so that the SFP level can be maintained.**

**NMC Response:**

The maximum calculated boil off rate of the Spent Fuel Pool is 42 gpm. Service Water, a seismically qualified QA-1 system and the emergency makeup source to the SFP, is capable of providing approximately 1000 gpm makeup water, much greater than the maximum boil off rate of 42 gpm.

49. **In Attachment 4, Section 4.2.4.1.1, "Main Steam System, Steam Generator Safety Valves," it is stated that:**

**The KNPP has 10 safety valves with a total capacity of 7.66 x $10^6$ lbs/hr, which provide about 107.3 percent of the current maximum design full-load steam flow of 7.14 x $10^6$ lb/hr. Based on the proposed range of NSSS design parameters approved for the power uprate, the installed safety valves provide about 98.6 percent of the maximum design steam flow of 7.77 x $10^6$ lbs/hr.**

**Further, in the same section, it is stated that:**

**The plant safety analysis for the power uprate presented in Section 6.2, which summarizes non-LOCA event analysis documented in RTSR confirms that the installed safety valve capacity of 7.66 x $10^6$ lbs/hr is adequate for overpressure protection.**

**Please expand on the above statements and provide clarification/justification on the adequacy for overpressure protection.**

**NMC Response:**

The KNPP design basis accident and transients were re-analyzed at the power uprate conditions. The results are documented in the Reload Transition Safety Report (RTSR) and in the power uprate licensing report (attachment 4 of our submittal). The results of the safety analyses demonstrate that the maximum main steam system (MSS) pressure, in all safety analysis cases, is less than the MSS pressure acceptance criteria of 110% of MSS design pressure.

The loss of load / turbine trip (LOL/TT) transient is the limiting transient for MSS pressure. The adequacy of the main steam safety valves (MSSVs) is demonstrated via the analysis of the LOL/TT transient, which is documented in the RTSR. Assumptions that maximize the resultant peak main steam system (MSS) pressure were applied in a LOL/TT case analyzed specifically for the MSS overpressure concern. The peak calculated MSS pressure is 1202 psia, which is less than the applicable acceptance

criterion limit of 1210 psia. Showing that this acceptance criterion is satisfied for the LOL/TT transient demonstrates the adequacy of the pressure-relieving capability of the MSSVs.

Through the design basis safety analyses at power uprate conditions, the capability and the capacity of the MSSV's are shown to meet their design basis safety function, and are confirmed adequate for overpressure protection of the MSS since the MSS pressure remains less that 110% of design pressure during all postulated design basis accidents and transients. The fact that, at the current licensed power level, the MSSV's have a greater flow capacity than the MS flow at 100% power, and at power uprate conditions the MSSV's have a capacity that is slightly less than the MSS steam flow at 100% power is not a safety or design concern. The capacity of the MSSV's being greater than or equal to the full power MSS steam flow is not a design basis criteria for the MSSV's.

**Questions from the EEIB - Instrumentation & Control Section**

50. **Please provide the calculations and supporting setpoint methodology document, WCAP-15821 used to determine the reactor trip setpoints given in WCAP-16040-P, Table 6.8-2. The detail should be sufficient to allow the staff to understand the values used, assumptions made, and formulae used.**

**NMC Response:**

WCAP-15821-P and –NP (proprietary and non-proprietary), Revision 0, Westinghouse Protection System Setpoint Methodology Kewaunee Nuclear Plant (Power Uprate to 1757 MWt-NSSS Power with Feedwater Venturis, or 1780 MWt-NSSS Power with Ultrasonic Flow Measurements, and 54F Replacement Steam Generators), are provided as Enclosures D and E.

Please note that the only Reactor Protection System trip setpoints being revised as a result of the power uprate are the Overtemperature ΔT (OTDT) and the Overpower ΔT (OPDT) setpoints. The rest of the Reactor Trip setpoints remain unchanged, and are per our current, approved Technical Specifications.

As described in Attachment 4 of our submittal, Section 6.8.2.1, and WCAP-15821-P (Enclosure D), Section 1.0, the Westinghouse Methodology utilized is consistent with Regulatory Guide 1.105, Setpoints for Safety-Related Instrumentation, and conforms to ISA Standard ANSI/ISA-67.04, Setpoints for Nuclear Safety-Related Instrumentation, which is a methodology accepted by the NRC per Regulatory Guide 1.105.

51. **Please provide KNPP General Nuclear Procedure 04.06.01, Plant Setpoint Accuracy Calculation Procedure.**

**NMC Response:**

Kewaunee procedure GNP-04.06.01, Revision E, Plant Setpoint Accuracy Calculation Procedure, is provided as Enclosure G. Additionally, a section that is a part of every I&C calculation, titled "Methodology", is provided as Enclosure H.

The setpoint calculation Methodology section details how the Instrument Loop Uncertainty Program (ILUP) software (referenced in GNP-04.06.01, steps 6.2.1.2 and 6.3.1.2) calculates loop and setpoint uncertainties. It also provides detail for the manual checking of an ILUP calculation, or the manual performance of a calculation.

As stated in both GNP-04.06.01 and the Methodology section of the I&C Calculations, the KNPP Plant Setpoint Accuracy program is based on ISA Standard ANSI/ISA-67.04, Setpoints for Nuclear Safety-Related Instrumentation. As discussed in NRC #50 above, Regulatory Guide 1.105, Setpoints for Safety-Related Instrumentation, finds the ISA standard an acceptable method for satisfying the NRC's regulations for maintaining safety-related setpoints.

52. **Please discuss the instrumentation and control recommendations mentioned in regulatory commitment 6 and NMC staff disposition.**

**NMC Response:**

The recommendations are still being evaluated as part of the Stretch Power Uprate modification process, but are required to be completed, with any required procedure and/or setpoint changes revised and ready for issue as part of the pre-requisites of the power uprate implementation plan. As stated in our LAR-195 submittal, Commitment 6, the recommendations will be reviewed, and implemented as appropriate, prior to implementation of the uprate.

## Questions from the Mechanical & Civil Engineering Branch

53. **In reference to Section 5.7.2, provide a summary of the results relating to the evaluation of SG, for the current rated and the power uprate conditions. The summary should include stresses, cumulative usage reactors (CUFs) and code allowables at limiting locations in the SG shell and the internal components including the manhole, U-bent tubes and divider plate. Also, provide an example to illustrate how you arrive at the calculated CUF value for the secondary side pressure boundary components following the power uprate.**

**NMC Response:**

A summary of the SG evaluation results, including stresses, cumulative usage factors (CUFs), and code allowables at the limiting locations in the SG (the limiting locations for structural integrity and fatigue analysis in the SG component are the feedwater nozzle and thermal sleeve and the J-nozzle to feedring weld) is provided in attachment 4 to our submittal, Section 5.7, Steam Generator Component Evaluations, subsection 5.7.2, Structural Integrity Evaluation and Tables 5.7-2, 5.7-3, and 5.7-4. These tables compare stress intensity analysis results between the current rated conditions, the power uprate conditions, and the ASME code limit. The limiting locations, feedwater nozzle and thermal sleeve and J -nozzle to feedring weld, bound all other locations in the SG component with respect to structural integrity and fatigue analysis. CUFs and detailed stress summary results for the critical components are provided in Attachment 4, section 5.7.2.4. Also included in section 5.7.2.2 and 5.7.2.4 of Attachment 4 is a discussion on

how the analysis was performed based on the use of temperature differences and scaling factors.

Detailed analysis of each individual component in the steam generators was not performed. For a power uprate structural integrity and fatigue analysis evaluation, the existing design basis for the SG component is reviewed and evaluated. The evaluation identifies the most highly stressed, and/or highest CUF locations of the SG. If those locations or parts remain qualified under the uprated power conditions, it is concluded that all other locations and parts of the steam generators will also be qualified, and will be adequate for the power uprate operating conditions. This approach is consistent with the standard approach for power uprate evaluations which has been accepted as a valid approach on all of the Uprated Power Licensing Amendments in which Westinghouse has been involved.

54. **In Section 8.4 of the submittal, you stated that an assessment of the BOP piping and supports (including main steam, condensate and feedwater, auxiliary feedwater and SG blowdown systems piping, etc.) was performed for an power uprate at 1772 MWt. You concluded that the piping and pipe supports remain in compliance with the USAS B31.1, "Power Piping Code."**

**In page 8-95 of your submittal, you concluded that the existing main steam piping remains acceptable for the power uprate condition based on the results of analysis with the higher flow rate resulting from the power uprate. Provide a summary of stresses, CUFs and code allowable limits for the critical locations in the main steam piping and support system at the current rated and the power uprate conditions.**

**NMC Response:**

The KNPP BOP system Code of Record is USAS B31.1, Power Piping Code, 1967. A detailed fatigue evaluation is not required for USAS B31.1. Based on a Main Steam (MS) System analysis with the higher flow rate resulting from the power uprate, MS system stress levels and allowables are included directly below.

**Table 1 - Maximum Pipe Stress Levels and Allowables for Main Steam**

| **Criteria** | **Allowable stress (psi)** | **Baseline** (1650 MWt) | **Max stress** (Power Uprate) |
|---|---|---|---|
| Criteria 5 | 1.2 Sh = 21000 psi | 20509 psi | 20665 psi |
| Criteria 6 | 1.8 Sh = 31500 psi | 23666 psi | 23837 psi |
| Criteria 5 (Upset): Pressure + Weight + SRSS (OBE Earthquake + Fluid transients due to Main Steam TSV event) | | | |
| Criteria 6 (Faulted): Pressure + Weight + SRSS (DBE Earthquake + Fluid transients due to MS turbine stop valve closure event) | | | |

55. **In reference to Section 8.4, provide a technical basis for not evaluating the piping and support systems where the increase in temperature, pressure and flow rate are less than 5 percent of the current rated design-basis condition. Your justifications provided on page 8-94 are qualitative and nonspecific. For instance, you stated that these increases are somewhat offset by conservatism in analytical methods used. You also indicated that conservatism may include the enveloping**

**of multiple thermal operating conditions. We can not draw a conclusion from these undefined qualitative statements. The technical justifications should be based on specific quantitative assessments or intuitively conservative deduction in order for us to accept your conclusions.**

**NMC Response:**

Main Steam is discussed in question #49 above. The Service Water and Component Cooling Water Systems require resolution of open items as identified in Attachment 7 of our submittal, Commitment #7. These items will be resolved prior to stretch uprate implementation. Therefore, this response will only address the Bleed Steam (also known as Extraction Steam), Condensate, Feedwater, and Heater Drain piping systems. All other BOP piping systems are unaffected by the SUR and have change factors ≤ 1.0. Therefore, further evaluation is not required.

Additional specific quantitative assessments were performed for the condensate, feedwater, bleed steam and heater drain piping systems to further support the previous conclusions that these piping systems remain acceptable for power uprate conditions. These assessments involved a review of the existing design basis pipe stress analyses for each piping system to determine the portions of each system that currently have the highest stress levels that will be impacted by the minor operating increases due to power uprate.

Table 1 provides the existing stress, the power uprate stress, and applicable allowable stress for critical locations (high pipe stress locations resulting in lowest design margin) for those portions of the subject piping systems that experienced an increase in operating temperature as a result of power uprate.

Table 1 – Summary of Pipe Stress Levels

| Piping System | Loading Condition | Existing Stress (psi) | Power Uprate Stress (psi) | Allowable Stress (psi) | Comments |
|---|---|---|---|---|---|
| Condensate | Sustained + Thermal | 27,986 | 28,434 | 37,500 | See Note 1 |
| Feedwater | Thermal | 12,403 | 12,527 | 22,500 | See Note 2 |
| Bleed Steam | Thermal | 18,065 | 18,246 | 22,500 | See Note 3 |
| Heater Drains | Thermal | 14,907 | 15,354 | 22,500 | See Note 4 |

Note 1
The stress data shown is for the condensate piping running between Heaters 14A&B and the FW pumps which experiences a temperature increase from 360°F to 366°F.

Note 2
The stress data shown is for the outside containment feedwater piping located downstream of Heaters 15A&B which experiences a temperature increase from 432°F to 437°F.

Note 3
The stress data shown is for the bleed steam piping running from the HP turbine to Heaters 14A&B which experiences a temperature increase from 365°F to 368°F.

Note 4
The stress data shown is for the heater drain piping running from Heaters 14A&B to the Heater Drain Tank which experiences a temperature increase from 360°F to 368°F.

The data summarized in Table 1 shows that the minor stress increases resulting from power uprate remain within the piping system allowable stress limits.

56. **In reference to Section 8.4, provide a summary of your evaluation for the BOP piping and supports including calculated maximum stresses and CUFs at critical locations of each evaluated piping system for the power uprate condition, code allowable limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide your justification.**

**NMC Response:**

See responses to questions #54 and #55 above.

57. **In reference to Section 8.7, discuss the functionality of safety-related mechanical components (i.e., all safety-related valves and pumps, including air-operated and power-operated valves) affected by the power uprate to ensure that the performance specifications and TS requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that safety-related motor-operated valves (MOVs) in your Generic Letter (GL) 89-10 MOV program at KNPP will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify mechanical components for which functionality at the uprated power level could not be confirmed. Also, discuss effects of the proposed power uprate on the**

**pressure locking and thermal binding of safety-related power-operated gate valves for GL 95-07 and on the evaluation of overpressurization of isolated piping segments for GL 96-06.**

**NMC Response:**

The MOV Program assumed design fluid system parameters for flows, temperatures, and differential pressures, which bound the uprated flows, temperatures, and differential pressures.

MOVs in the Plant MOV Valve Program were reviewed for impact resulting from uprated post accident environmental conditions. Valves inside containment utilized design post accident environmental conditions which bound the uprated power conditions and are not impacted. Degraded voltage and thrust calculations for MOV operators outside containment were reviewed for impact of the uprated post accident temperatures, and will be revised, as required, prior to implementation of the 6% Stretch Power Uprate (this will be new Regulatory Commitment #16 in Attachment 6 of this submittal).

The ISI Program is not impacted by the 6% Stretch Uprate. KNPP recently implemented a Risk Informed ISI program for Class 1 and 2 piping. When this program was initiated, the predicted system and component parameters for a 7.4% uprate were incorporated into the program.

The IST Program is not impacted since the power uprate does not add or change the design function or system performance requirements of any safety related systems or components.

The Plant AOV and Check Valve Programs were determined not to be impacted. These programs use system design capacities for flows, temperatures, and differential pressures, which bound the uprated flows, temperatures, and differential pressures across safety related valves.

Issues such as the valve thermal binding and pressure locking (GL 95-07) and thermally induced over pressurization of water filled piping inside containment (GL 96-06) were determined not to be impacted, because the new post uprate conditions are bounded by existing program boundary conditions.

ATTACHMENT 2

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

**November 5, 2003**

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Responses to Requests for Additional Information and Supplemental Information Regarding LAR 195

TS Bases pages TS B3.3-3 and TS B4.8-1 marked up to show the proposed changes

2 Pages to Follow

The containment cooling function is provided by two systems: containment fancoil units and containment spray systems. The containment fancoil units and containment spray system protect containment integrity by limiting the temperature and pressure that could be experienced following a Design Basis Accident. The Limiting Design Basis accidents relative to containment integrity are the loss-of-coolant accident and steam line break. During normal operation, the fancoil units are required to remove heat lost from equipment and piping within the containment.[2] In the event of the Design Basis Accident, ~~any one~~either of the following combinations will provide sufficient cooling to limit containment pressure to less than design values: four fancoil units~~, two containment spray pumps,~~ or two fancoil units plus one containment spray pump.[3]

In addition to heat removal, the containment spray system is also effective in scrubbing fission products from the containment atmosphere. Therefore, a minimum of one train of containment spray is required to remain OPERABLE in order to scavenge iodine fission products from the containment atmosphere and ensure their retention in the containment sump water.[4] [5]

Sodium Hydroxide (NaOH) is added to the spray solution for pH adjustment by means of the spray additive system. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the containment sump.

The alkaline pH of the containment sump water inhibits the volatility of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the sump fluid. Test data has shown that no significant stress corrosion cracking will occur provided the pH is adjusted within 2 days following the Design Basis Accident.[6] [7]

A minimum of 300 gallons of not less than 30% by weight of NaOH solution is sufficient to adjust the pH of the spray solution adequately. The additive will still be considered available whether it is contained in the spray additive tank or the containment spray system piping and Refueling Water Storage Tank due to an inadvertent opening of the spray additive valves (CI-1001A and CI-1001B).

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[2] USAR Section 6.3
[3] USAR Section 6.4
[4] USAR Section 6.4.3
[5] USAR Section 14.3.5
[6] USAR Section 6.4
[7] Westinghouse Chemistry Manual SIP 5-1, Rev. 2, dated 3/77, Section 4.

**BASIS**

The Auxiliary Feedwater System (AFW) mitigates the consequences of any event that causes a loss of normal feedwater. The design basis of the AFW System is to remove decay and residual heat by delivering the minimum required flow to at least one steam generator until the Reactor Coolant System (RCS) is cooled to the point of placing the Residual Heat Removal System into operation.

In accordance with ASME Code Section XI, Subsection IWP, an in-service test of each auxiliary feedwater pump shall be run nominally every 3 months (quarterly) during normal plant operation. It is recommended that this test frequency be maintained during shutdown periods if this can be reasonably accomplished, although this is not mandatory. If the normally scheduled test is not performed during a plant shutdown, then the motor-driven pumps shall be demonstrated OPERABLE within 1 week exceeding 350°F; and the turbine-driven pump shall be demonstrated OPERABLE within 72 hours of exceeding 350°.

Quarterly testing of the AFW pumps is used to detect degradation of the component. This type of testing may be accomplished by measuring the pump's developed head at one point of the pump characteristic curve. This verifies that the measured performance is within an acceptable tolerance of the original pump baseline performance.

TS 3.4.b requires all three AFW pumps be OPERABLE prior to heating the RCS average temperature > 350°F. It is acceptable to heat the RCS to > 350°F with the turbine-driven pump inoperable for a limited time period of 72 hours. The wording of TS 3.4.b.<u>4</u>~~2~~.B and TS 4.8.b allows delaying the testing until the steam flow is consistent with the conditions under which the performance acceptance criteria were generated.

The discharge valves of the two motor-operated pumps are normally open, as are the suction valves from the condensate storage tanks and the two valves on a cross tie line that directs the turbine-driven pump discharge to either or both steam generators. The only valve required to function upon initiation of auxiliary feedwater flow is the steam admission valve on the turbine-driven pump. Proper opening of the steam admission valve will be demonstrated each time the turbine-driven pump is tested.

ATTACHMENT 3

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

**November 5, 2003**

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Responses to Requests for Additional Information and Supplemental Information Regarding LAR 195

Revised (clean copy) TS Bases pages TS B3.3-3 and TS B4.8-1

2 Pages to Follow

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A minimum of 300 gallons of not less than 30% by weight of NaOH solution is sufficient to adjust the pH of the spray solution adequately. The additive will still be considered available whether it is contained in the spray additive tank or the containment spray system piping and Refueling Water Storage Tank due to an inadvertent opening of the spray additive valves (CI-1001A and CI-1001B).

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(2) USAR Section 6.3
(3) USAR Section 6.4
(4) USAR Section 6.4.3
(5) USAR Section 14.3.5
(6) USAR Section 6.4
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The discharge valves of the two motor-operated pumps are normally open, as are the suction valves from the condensate storage tanks and the two valves on a cross tie line that directs the turbine-driven pump discharge to either or both steam generators. The only valve required to function upon initiation of auxiliary feedwater flow is the steam admission valve on the turbine-driven pump. Proper opening of the steam admission valve will be demonstrated each time the turbine-driven pump is tested.

ATTACHMENT 4

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

**November 5, 2003**

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Responses to Requests for Additional Information and Supplemental Information Regarding LAR 195

Supplement to Containment Integrity Safety Analysis

26 Pages to Follow

Introduction and Background

By application dated September 2, 2002 (reference 2) as supplemented by letters dated July 23 and September 26, 2003, the Nuclear Management Company (NMC) requested changes to the Kewaunee Nuclear Power Plant (KNPP) analysis-licensing basis. Specifically, the proposed changes would revise the licensing basis from GOTHIC 6.0 (version 6.0a) to GOTHIC 7.0 (version 7.0p2) for containment integrity analyses.

The Nuclear Regulatory Commission (NRC) had previously approved the use of the GOTHIC 6.0 computer code for the calculation of containment response to design-basis loss-of-coolant accidents (LOCA) and the main steam line break accidents (Reference 5). NMC stated in reference 2 that GOTHIC 7.0 would be used for the same purposes that were noted in the GOTHIC 6.0 approval. NMC also stated that the principal difference between GOTHIC 6.0 and GOTHIC 7.0 is a mist diffusion layer model (MDLM), although several other changes were discussed.

The NRC authorized (Reference 3) the use of the upgraded computer code for design-basis accident containment integrity analyses called Generation of Thermal-Hydraulic Information for Containment (GOTHIC) version 7.0p2 (GOTHIC 7) with the following conditions:

1. The height effect scaling factor $\lambda_h$ applied to the heat and mass transfer analogy shall not be used for the Kewaunee licensing calculations.
2. The Gido-Koestel (G-K) correlation shall not be used for Kewaunee licensing calculations.
3. The inclusion of mist in the mist diffusion layer model (MDLM) shall not be used for Kewaunee licensing calculations.

In addition,

4. It is not necessary to apply the proposed bias term to the mist diffusion layer model for Kewaunee licensing calculations.
5. It is not necessary to use a combination of Uchida and MDLM for the containment heat structures. MDLM may be used for heat transfer to all structures for Kewaunee licensing calculations.

The restrictions imposed on the MDLM heat and mass transfer option require some GOTHIC code modifications. These modifications were implemented in GOTHIC version 7.1 Patch1, released under Numerical Applications Incorporated's (NAI's) QA Program. GOTHIC version 7.1 Patch1 includes the modifications to implement the DLM heat and mass transfer option to satisfy the NRC restrictions on the MDLM option. Under the DLM option there is no enhancement of the heat or mass transfer due to mist generation in the thermal boundary layer. The other conditions imposed by the NRC on the heat and mass transfer model are implemented by code input. The modifications to create version 7.1Patch1 from version 7.1 are described in the 7.1Patch1 Release Notes. Other than the DLM heat and mass transfer option, none of the changes implemented in version 7.1 Patch1 will have any impact on the results of the analyses previously submitted for KNPP containment integrity analyses (CIA) in reference 4, attachment 4, section 6.4.

To confirm that the code differences between GOTHIC 7.0p2 and 7.1 Patch1 have no significant, unintended impact on the CIA, the original 7.0p2 containment evaluation models were re-run using version 7.1 Patch1 without making any model changes. Results show no

impact on the KNPP CIA. The containment evaluation models were then modified to implement the DLM heat and mass transfer option and the revised heat sinks.

The following containment evaluation model changes are implemented to comply with the restrictions from the NRC approval of the MDLM heat transfer option (reference 3).

a) The new DLM option is used for all conductors exposed to the vapor, including those that previously used the Uchida option. The new DLM option is the same as the MDLM option except that there is no mist formation in the boundary layer and no associated enhancement of the heat and mass transfer.

b) The characteristic height parameter for all conductors using the new DLM option is set to 1 ft. This ensures that there will be no enhancement of the heat and mass due to film roughening effects.

c) The previously imposed multipliers to conservatively bias the DLM heat and mass transfer option are removed.

d) Heat Sinks are revised. In the MSLB containment evaluation model, the heat loss to the shield building through the containment shell is included and the accumulators are explicitly modeled. For MSLB and LOCA containment evaluation models the surface area and thickness of the conduits and cable trays are increased to be consistent with more recent plant inventory data. For the LOCA, only the heat sink for the conduits and cable trays is revised. The accumulators are not in the existing LOCA containment evaluation model and are not included in the new models. The LOCA cases are used for the long term cooling analysis and the simple shield building model used in the MSLB cases is not appropriate for the LOCA cases.

The GOTHIC computer code was also used to analyze the High Energy Line Break (HELB) outside of containment. HELB outside the reactor containment structure may result in significant releases of high-energy fluid to the systems, structures, and components surrounding these high- energy systems. The structural loads associated with HELB outside containment are compartment differential pressures, temperature transients, and the static and dynamic effects of pipe rupture restraint reactions, jet impingement loads, and pipe whip loads. Structural loads for power uprate were evaluated and documented in section 8.5 of attachment 4 of reference 4. In addition to the structural loads, the high-energy line break outside containment may result in severe environmental conditions, high pressure and high temperature conditions, used for equipment qualification (EQ).

The HELB compartment analysis for power uprate was performed with the GOTHIC code. A GOTHIC (version 6.0) auxiliary building compartment model was developed and used for the replacement steam generator (RSG) HELB analyses. The GOTHIC 7.0 code version was benchmarked to the GOTHIC 6.0 RSG results and achieved excellent agreement. The GOTHIC KNPP auxiliary building model with GOTHIC code version 7.0, without the MDLM model, was applied to the power uprate HELB compartment analyses. In addition, the KNPP GOTHIC auxiliary building model with GOTHIC code version 7.1, without the MDLM option, is being used to perform thermal-lag analyses of certain EQ components outside containment.

Since the outside containment power uprate analyses do not use the MDLM model for the HELB compartment analyses these analyses using the GOTHIC 7.0 or GOTHIC 7.1 code version are fully compliant with the reference 5 Safety Evaluation (SE) and MDLM heat and mass transfer restrictions.

## Loss-of-Coolant Accident Containment Response Analysis

The KNPP containment system is designed so that for all LOCA break sizes, up to and including the double-ended severance of a reactor coolant pipe, the containment peak pressure remains below the design pressure. This section details the containment response subsequent to a hypothetical LOCA. The containment response analysis uses the long-term LOCA mass and energy release data from subsection 6.4.1.1 in attachment 4 to reference 4.

The containment response analysis demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a LOCA inside containment. The impact of LOCA mass and energy releases on the containment pressure is addressed to ensure that the containment pressure remains below its design pressure at the licensed core power conditions. In support of equipment design and licensing criteria (for example, qualified operating life), with respect to post-accident environmental conditions, long-term containment pressure and temperature transients are generated to conservatively bound the potential post-LOCA containment conditions.

### Accident Description

A break in the primary RCS piping causes a loss-of-coolant, which results in a rapid release of mass and energy to the containment atmosphere. Typically the blowdown phase for the large LOCA events (DEHL, cold leg, or pump suction pipe breaks) is over in less than 30 seconds. This large and rapid release of high-energy, two-phase fluid causes a rapid increase in the containment pressure, which results in the actuation of the emergency fan cooler and containment spray systems.

The RCS accumulators begin to refill the lower plenum and downcomer of the reactor vessel with water after the end of blowdown. The reflood phase begins after the vessel fluid level reaches the bottom of the fuel. During this phase, the core is quenched with water from both the accumulators and pumped SI. The quenching process creates a large amount of steam and entrained water that is released to containment through the break. This two-phase mixture would have to pass through the steam generators and also absorb energy from the secondary side coolant if the break were located in the cold leg or pump suction piping.

The LOCA mass and energy release decreases with time as the system cools and depressurizes. Core decay heat is removed by nucleate boiling after the reflood phase is complete. The core fluid level is maintained by pumping water back into the vessel from either the SI or sump recirculation system. The containment heat removal systems continue to condense steam and slowly reduce the containment pressure and temperature over time.

### Input Parameters and Assumptions

A series of analyses, using different break sizes and locations, was performed for the LOCA containment response. Subsection 6.4.1.1 in attachment 4 to reference 4 documented the mass and energy releases for the DEPS and DEHL breaks. The DEPS break cases were run with both minimum and maximum safeguards. The three minimum safeguards cases assume a diesel train failure. This assumption leaves one of two containment spray pumps and two of four containment fan coil units (CFCUs) available for containment heat removal. Three variations on the RHR/CCW configuration were considered for the minimum safeguards case. The first case assumes the operation of one RHR heat exchanger with CCW flow of 1550 gpm, 1 CCW heat exchanger with SW flow and one CCW heat exchanger without SW flow. The second case assumes 1 RHR heat exchanger with CCW flow, 1 RHR heat exchanger without CCW flow, 1 CCW heat exchanger with SW flow and 1 CCW heat exchanger without SW flow. The third case is the same as the first except that the CCW flow to the RHR heat exchanger is

reduced to 1100 gpm. Two single-failure cases were modeled for the maximum safeguards DEPS case. In the first case, one of the two containment spray pumps was assumed to fail, and in the second case one of the four CFCUs was assumed to fail. Only one RHR heat exchanger was credited for recirculation cooling in the maximum safeguards DEPS cases.

The containment initial conditions (pressure, temperature, and humidity) assumed for the containment response analyses are shown in Table 1.

Table 1 also includes the temperature of the service water system (SWS), the initial temperature of the refueling water storage tank (RWST), and the containment cooling system assumptions used in the analysis.

The heat sink data for the Kewaunee containment model is summarized in Table 2. The thermo-physical properties of the containment heat sink materials are shown in Table 3. The CFCU performance data (heat removal as a function of containment temperature) is shown in Table 4.

The major assumptions made in the containment response analysis are listed below:

- The LOCA mass and energy release input to the containment model is described in subsection 6.4.1.1 in attachment 4 to reference 4.
- Homogeneous mixing is assumed. The steam-air mixture and the water phases each have uniform properties. More specifically, thermal equilibrium between the air and the steam is assumed. However, this does not imply thermal equilibrium between the steam-air mixture and the water phase.
- Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermo-dynamic properties.
- For the blowdown portion of the LOCA analysis, the discharge flow separates into steam and water phases at the breakpoint. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment. Steam and water releases are input separately for the post-blowdown portion of the LOCA analysis.
- The saturation temperature at the steam partial pressure is used for heat transfer to the heat sinks and the fan coolers.

**Description of the Kewaunee GOTHIC Containment Model**

Calculation of the containment pressure and temperature is accomplished by use of the digital computer code GOTHIC. GOTHIC version 7.1 Patch1 was used for this analysis.

The heat sink data for the Kewaunee containment model is summarized in Table 2. The thermo-physical properties of the containment heat sink materials are shown in Table 3.

An improved recirculation heat removal system model was added to the Kewaunee containment model to more accurately determine the RHR and CCWS temperatures during sump recirculation for the LOCA analysis. The containment peak pressure and temperature occur prior to the transfer to recirculation; the improved recirculation model only affects the long-term LOCA containment pressure and temperature response.

The recirculation system model uses GOTHIC component models for the RHR and CCW heat exchangers and the CCW pump. The GOTHIC heat exchanger input data was taken from the heat exchanger specification sheets. The heat exchanger models were benchmarked against design conditions and output data from the COCO code.

The RHR System model uses a flow boundary condition to draw suction from the sump through the RHR heat exchanger. The RHR heat exchanger transfers energy from the sump to the CCWS. The CCWS model calculates the secondary side inlet conditions for the RHR heat exchanger. The CCW pump provides flow through the CCW heat exchanger to transfer heat to the Service Water System (SWS). The service water flow rate and temperature are boundary conditions to the CCW heat exchanger model. The CCW heat exchanger outlet flow is split between the RHR heat exchanger and the other CCW heat loads. The other heat loads are modeled using a constant heat source.

**Acceptance Criteria**

The containment response for design-basis containment integrity is an American Nuclear Society (ANS) Condition IV event, an infrequent fault. The containment analysis methodology satisfies the current NRC acceptance criteria from 10CFR50, Appendix A and SRP 6.2.1.1.A. The relevant general design criteria (GDC) requirements that are met are as follows:

- GDC 16 and GDC 50: To satisfy the requirements of GDC 16 and GDC 50, the peak calculated containment pressure should be less than the containment design pressure of 46 psig considering the most severe single failure
- GDC 38 and GDC50: To satisfy the requirements of GDC 38 and GDC 50, the calculated pressure at 24 hours should be less than 50 percent of the peak calculated value. (This is related to the criteria for doses at 24 hours.)

The Kewaunee plant was originally licensed with the FSAR containing text from the interim criteria that was derived from the draft Atomic Industrial Forum (AIF) GDCs. The Kewaunee SER indicated that the operating license was granted because "...the plant design generally conforms to the intent...." of the requirements of 10CFR50 Appendix A. The specific interim criteria was:

- Interim GDC 10 - Containment
- Interim GDC 49 - Containment Design Basis
- Interim GDC 52 - Containment Heat Removal Systems

**Analysis Results**

The containment response calculations for the DEPS case with minimum Safety Injection was performed for 3 million seconds (approximately 35 days). The containment response calculations for the DEPS case with maximum Safety Injection was performed for 10 million seconds (approximately 116 days). Since the steam generator secondary side energy is effectively isolated for hot leg breaks, the containment response calculation for the DEHL case was performed for the blowdown phase only (approximately 20 seconds).

The containment pressure, steam temperature, and water (sump) temperature profiles from each of the LOCA cases are shown in Figures 1 through 12. Table 5 summarizes the LOCA containment response results for the three cases studied.

**Conclusions**

The LOCA containment response analyses have been performed as part of the Power Uprate Program for Kewaunee. The analyses included long-term pressure and temperature profiles for each case. The calculated peak containment pressure was less than the design pressure (46 psig) for all cases. In addition, the containment pressure was less than 50 percent of the peak value within 24 hours. Based on the results, all applicable containment integrity acceptance criteria for Kewaunee have been met.

MSLB Containment Response Analysis

The KNPP containment system is designed so that at any power level with any steam line break size up to, and including the double-ended severance of the main steam line, the containment peak pressure remains below the design pressure. This section details the containment response subsequent to a hypothetical main steam line break (MSLB). The containment response analysis uses the MSLB mass and energy release data from subsection 6.4.2.1 in attachment 4 to reference 4 and the mass and energy release data shown in table 6 of this supplement report.

The containment response analysis demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a MSLB inside containment. The impact of MSLB mass and energy releases on the containment pressure is addressed to ensure that the containment pressure remains below its design pressure at the uprated core power conditions.

**Accident Description**

A break in the main steam line piping causes a rapid release of secondary side mass and energy to the containment atmosphere. This release of high-energy steam causes the containment pressure to increase, which results in the actuation of the CFCUs and containment spray.

The MSLB mass and energy release decreases with time as the system cools and depressurizes. The containment heat removal systems continue to condense steam and slowly reduce the containment pressure and temperature over time.

**Input Parameters and Assumptions**

A series of analyses using different break sizes and power levels was performed for the MSLB containment response. Subsection 6.4.2.1 in attachment 4 to reference 4 documents the MSLB mass and energy release analyses. Single active failures were considered in these analyses. In one set of cases, the feedwater regulator valve (FRV) was assumed to fail, and in another set of cases the MSIV was assumed to fail. A third set of MSLB mass and energy release cases was run without any additional steam or feedwater system failures. For this set, a single failure of one of the diesel generators to start and load one of two trains of safety equipment is assumed. Under these conditions, only one of two spray pumps and two of four fan coolers provide containment cooling.

The containment initial conditions (pressure, temperature, and humidity) assumed for the containment response analyses are shown in Table 1.

Table 1 also includes the temperature of the service water system (SWS), the initial temperature of the refueling water storage tank (RWST), and the containment cooling system assumptions used in the analysis.

The heat sink data for the Kewaunee containment model is summarized in Table 2. The thermo-physical properties of the containment heat sink materials are shown in Table 3. The CFCU performance data (heat removal as a function of containment temperature) is shown in Table 4.

The major assumptions made in the containment response analysis are listed below:

- The MSLB mass and energy release input to the containment model is described in subsection 6.4.2.1 in attachment 4 to reference 4 and in table 6 of this supplement report.

- Homogeneous mixing is assumed. The steam-air mixture and the water phases each have uniform properties. More specifically, thermal equilibrium between the air and the steam is assumed. However, this does not imply thermal equilibrium between the steam-air mixture and the water phase.
- Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermo-dynamic properties.

**Description of the Kewaunee GOTHIC Containment Model**

The digital computer code GOTHIC was used to calculate the containment pressure and temperature. GOTHIC version 7.1Patch1 was used for this analysis.

The heat sink data for the Kewaunee containment model is summarized in Table 2. Additional heat sinks for the conduit and cable trays, accumulators and the containment shield building have been included. The thermo-physical properties of the containment heat sink materials are shown in Table 3.

**Acceptance Criteria**

The containment response for design-basis containment integrity is an ANS Condition IV event, an infrequent fault. The containment analysis methodology satisfies the current NRC acceptance criteria from 10CFR50, Appendix A and Standard Review Plan 6.2.1.1.A. The relevant GDC requirements that are met are as follows:

- GDC 16 and GDC 50: To satisfy the requirements of GDC 16 and GDC 50, the peak calculated containment pressure should be less than the containment design pressure of 46 psig, considering the most severe single failure.
- GDC 38 and GDC 50: To satisfy the requirements of GDC 38 and GDC 50, the calculated pressure at 24 hours should be less than 50 percent of the peak calculated value. (This is related to the criteria for doses at 24 hours.)

The Kewaunee plant was originally licensed with the FSAR containing text from the interim criteria that was derived from the draft AIF GDCs. The Kewaunee SER indicated that the operating license was granted because "...the plant design generally conforms to the intent...." of the requirements of 10CFR50, Appendix A. The specific interim criteria were:

- Interim GDC 10 - Containment
- Interim GDC 49 - Containment Design Basis
- Interim GDC 52 - Containment Heat Removal Systems

**Analysis Results**

The containment evaluation model and assumptions described in subsections 6.4.2.2.2 and 6.4.2.2.3 of reference 4, attachment 4 and as updated by this supplement report, were used to determine the accident progression and containment response to the MSLB event.

A number of MSLB cases covering various break sizes and power levels were run using the GOTHIC containment model to determine the limiting case. Table 7 summarizes the peak containment pressures and temperatures calculated for these cases. The limiting MSLB event represents a 1.4 $ft^2$ break at 0-percent power conditions. Offsite power is available, but a single failure of one train of containment safeguards is assumed.

The containment pressure and temperature response for the limiting MSLB case is shown in Figures 13 and 14. The containment pressure and temperature increase steadily in response to

the steam release. Operator action to terminate AFW flow to the faulted steam generator is assumed to occur at 600 seconds. This significantly reduces the break flow rate from the faulted steam generator causing the containment pressure and temperature to begin decreasing. The containment pressure and temperature will continue to decrease since the break steam release is much less than the heat removal capability of the CFCUs and containment spray. The calculated peak containment pressure and temperature for this event is 45.7 psig and 266.6 F respectively.

**Conclusions**

The MSLB containment response analyses have been performed as part of the Power Uprate Program for Kewaunee. The calculated peak containment pressure was less than the design pressure (46 psig) for all cases. In addition, the containment pressure was less than 50 percent of the peak value within 24 hours. Based on the results, all applicable containment integrity acceptance criteria for Kewaunee have been met.

References

1. R. Ofstun, Development and Qualification of a GOTHIC Containment Evaluation Model for the Kewaunee Nuclear Power Plant, WCAP-15427, Rev. 1, April 2001.
2. T. Coutu, NMC - NRC-02-082, Kewaunee Nuclear Power Plant Request for Use of GOTHIC 7 in Containment Design Basis Accident Analyses, to NRC Document Control Desk, September 2, 2002.
3. NRC letter to Mr. Thomas Coutu, "Kewaunee Nuclear Power Plant – Issuance of Amendment (TAC No. MB6408)," dated September 29, 2003 with attached Safety Evaluation, Docket No. 50-305.
4. KNPP Letter #NRC-03-057 dated 5/22/03, License Amendment Request (LAR) 195 for Stretch Power Uprate
5. JB Lamb (NRC) to ME Reddemann (NMC), transmitting the NRC SER approving WPSRSEM-NP Revision 3, "Kewaunee Nuclear Power Plant - Review for Kewaunee Reload Safety Evaluation Methods Topical Report," Letter No. K-01-112, dated September 10, 2001

**Table 1**

**Containment Response Analysis Parameters**

| | |
|---|---|
| Service Water Temperature (°F) | 80 (0-24 hrs)<br>73 (24-168 hrs)<br>70 (>168 hrs) |
| RWST Water Temperature (°F) | 120 |
| Initial Containment Temperature (°F) | 120 |
| Initial Containment Pressure (psia) | 16.85 |
| Initial Relative Humidity (%) | 17.7 |
| Net Free Volume (ft3) | $1.32 \times 10^6$ |
| **CFCU** | |
| Total | 4 |
| Analysis Maximum | 4 |
| Analysis Minimum | 2 |
| Containment High Pressure Setpoint (psig) | 5.00 |
| Delay Time (sec)<br>With Offsite Power<br>Without Offsite Power | <br>75.0<br>85.0 |
| **Containment Spray Pumps** | |
| Total | 2 |
| Analysis Maximum | 2 |
| Analysis Minimum | 1 |
| Flow Rate (gpm)<br>Injection Phase (per pump)<br>Recirculation Phase | <br>1170<br>Not modeled |
| Containment High-High Pressure Setpoint (psig) | 23.0 |
| Delay time (sec)<br>With Offsite Power (delay after high-high pressure setpoint)<br>Without Offsite Power (delay after high-high pressure setpoint) | <br>106.0<br>135.0 |
| | |
| CS Termination Time, (sec)<br>Minimum Safeguards<br>Maximum Safeguards (stop one pump/stop second pump) | <br>3953<br>1253/2707 |

| Table 1 (Cont.)<br>Containment Response Analysis Parameters | |
|---|---|
| **RHR System** | |
| Recirculation Switchover, Full Flow Established, (sec)<br>Minimum Safeguards<br>Maximum Safeguards | <br>6382<br>1473 |
| Number of Heat Exchangers Modeled in the Analysis | 1 |
| RHR Flows through RHR Heat Exchangers | |
| Minimum Safeguards | |
| Time (sec) | Flow (lbm/s) |
| 0.0 | 0.0 |
| 4143 | 0.0 |
| 4143.1 | 33.3 |
| 6382 | 33.3 |
| 6382.1 | 186.2 |
| 3.1E+6 | 186.2 |
| Maximum Safeguards | |
| Time (sec) | Flow (lbm/s) |
| 0.0 | 0.0 |
| 1472 | 0.0 |
| 1473 | 186.2 |
| 3.1E+6 | 186.2 |
| CCW Flow per RHR Heat Exchanger (GPM) | 1000 |
| **CCW Heat Exchangers** | |
| Number of Heat Exchangers Modeled in the Analysis | 1 |
| CCW Flow per CCW Heat Exchanger (GPM) | 1600 |
| Service Water Flow (GPM) | 2000 |
| Additional Heat Loads, Btu/hr | $9.6 \times 10^6$ |

**Table 2**

**Kewaunee Structural Heat Sinks for Containment Integrity Analysis[1,2,3]**

| Sink | Surfaces Description | Material | Total Exposed Area ($ft^2$) | Thickness (in) |
|---|---|---|---|---|
| 1 | Containment Cylinder – Coating #4 | Carbon Steel | 41,300 | 1.5 |
| 2 | Containment Dome - Coating #4 | Carbon Steel | 17,300 | 0.75 |
| 3 | Reactor Vessel Liner – Coating #4 | Carbon Steel - Concrete Backup | 1260<br>1260 | 0.25<br>12.00 |
| 4 | Refueling Canal | Stainless Steel - Concrete Backup | 1100<br>1100 | 0.1875<br>12.0 |
| 5 | Refueling Canal | Stainless Steel - Concrete Backup | 5500<br>5500 | 0.25<br>12.0 |
| 6 | Misc. Supports – Coating #4 | Carbon Steel | 4055 | 0.168 |
| 7 | Misc. Supports – Coating #4 | Carbon Steel | 16,925 | 0.25 |
| 8 | Misc. Supports – Coating #4 | Carbon Steel | 28,500 | 0.375 |
| 9 | Crane – Coating #5 | Carbon Steel | 2000 | 0.75 |
| 10 | Crane – Coating #5 | Carbon Steel | 500 | 1.0 |
| 11 | Hand Rails – Coating #4 | Carbon Steel | 1695 | 0.0725 |
| 12 | Grating – Coating #4 | Carbon Steel | 12,400 | 0.045 |
| 13 | Exposed Conduit and Cable Trays – Coating #4 | Carbon Steel | 10,733 | 0.10 |
| 14 | Ductwork – Coating #4 | Carbon Steel | 18,000 | 0.035 |
| 15 | Walls 1' to 1.9' – Exposed 2 sides – Coating #2 | Concrete | 2806 | 6.0 |

| Table 2 (cont) Kewaunee Structural Heat Sinks for Containment Integrity Analysis[1,2,3] | | | | |
|---|---|---|---|---|
| 16 | Floors 12.0 in and Greater – Coating #2 | Concrete | 12,896 | 12.0 |
| 17 | Walls 4' to 7' 4" – Exposed 2 Sides – Coating #2 | Concrete | 18,588 | 24.0 |
| 18 | Floor (in contact with sump) – Coating #2 | Concrete | 1088 | 12.0 |
| 19 | Walls 2' to 3' 2" – Exposed 2 Sides – Coating #2 | Concrete | 28,898 | 12.0 |
| 20 | Floors 4 in to 10 in – Coating #2 | Concrete | 6810 | 4.0 |
| 21 | Accumulator Upper Dome Coating #4 | Carbon Steel Stainless Steel Liner | 307 | 1.375 |
| 22 | Accumulator Upper Cylinder Coating #4 | Carbon Steel Stainless Steel Liner | 494 | 2.75 |
| 23 | Accumulator Lower Cylinder Coating #4 | Carbon Steel Stainless Steel Liner | 869 | 2.75 |
| 24 | Accumulator Lower Dome Coating #4 | Carbon Steel Stainless Steel Liner | 307 | 1.375 |

Notes:

1. The accumulator conductors are included for the MSLB models only. A separate volume is modeled for the water and nitrogen in the accumulators. The inside of the accumulator upper dome and cylinder are in contact with the nitrogen in the accumulators with a natural convection heat transfer coefficient. The inside of the accumulator lower dome and cylinder are in contact with the water in the accumulators with a natural convection heat transfer coefficient. The assumed water volume in each accumulator is 1,225 gallons (minimum volume).
2. Using 11mil paint thickness from CONTEMPT model (KLIC-99-008).
3. There is an air annulus (shield building) between the concrete containment cylinder and dome and steel shell. For the MSLB models, a separate volume was included for the shield building with natural convection to the outside surface of the containment dome and cylinder.

PAINT COATING SYSTEMS:

Coating #1: Plastite 9028 surfacer – flush: Phenoline 305 Primer – 4 mils; Phenoline 305 Finish – 4 mils

Coating #2: Plastite 9028 Amine-Epoxy Filler – flush: Plastite 9009 Primer – 6 mils; Phenoline 300 Finish – 8 mils

Coating #3: Carbozinc 11 Primer – 3 mils; Phenoline 305 Finish – 4 mils

Coating #4: Carbozinc 11 Primer – 3 mils; Phenoline 305 Finish – 8 mils

Coating #5: Carbozinc 11 Primer – 3 mils

1 mil = 1/1000 inch

**Table 3**

**Thermo-Physical Properties of Containment Heat Sinks[1]**

| Material | Conductivity (Btu/hr-ft-°F) | Volumetric Heat Capacity (Btu/ft$^3$-°F) |
|---|---|---|
| Carbon Steel | 26.0 | 56.4 |
| Stainless Steel | 8.0 | 56.6 |
| Concrete | 0.80 | 28.8 |
| Phenoline 300 Finish | 0.25 | 32.4 |
| Phenoline 305 Finish | 0.25 | 32.4 |
| Phenoline 305 Primer | 0.25 | 32.4 |
| Carbozinc 11 Primer | 0.9 | 28.8 |

Notes:

1. For the LOCA cases more conservative values of 0.083 and 28.8 for the conductivity and heat capacity, respectively.

**Table 4**

**CFCU Performance**

| Containment Temperature (°F) | Heat Removal Rate (Btu/sec) per CFCU |
|---|---|
| 100 | 0 |
| 136 | 1858.3 |
| 205 | 8338.9 |
| 244 | 12691.7 |
| 270 | 15230.6 |
| 300 | 15230.6 |

**Table 5**

**LOCA Containment Response Results (loss-of-offsite power assumed)**

| Case | Peak Press. (psig) | Peak Temp. (°F) | Pressure (psig) @ 24 hours | Temperature (°F) @ 24 hours |
|---|---|---|---|---|
| DEPSMINSI | 43.1 @ 58.2 sec | 261.6 @ 38.2 sec | 10.1 @ 86,400 sec | 167.1 @ 86,400 sec |
| DEPSMAXSI<br>1 Fan Cooler Fails | 42.6 @ 58.1 sec | 261.7 @ 38.0 sec | 8.2 @ 86,400 sec | 153.4 @ 86,400 sec |
| DEPSMAXSI<br>1 Spray Pump Fails | 42.6@ 58.1 sec | 261.7 @ 38.0 sec | 7.1 @ 86,400 sec | 145.4 @ 86,400 sec |
| DEHL | 44.6 @ 19.9 sec | 265.0 @ 19.8 sec | Not applicable | Not applicable |

**Table 6**

**Main Steam line Break Inside Containment Mass and Energy Release Results for Limiting Case (1.4ft$^2$ Break at 0% Power)**

| Time (sec) | Break Flow (lbm/sec) | Break Enthalpy (Btu/lbm) |
|---|---|---|
| 0.00 | 0.0 | 0.0 |
| 0.15 | 5832.3 | 1192.6 |
| 0.65 | 5621.4 | 1193.9 |
| 1.15 | 5456.2 | 1185.7 |
| 1.65 | 5466.5 | 1153.2 |
| 2.15 | 5525.2 | 1116.4 |
| 2.65 | 5635.9 | 1073.6 |
| 3.15 | 5734.3 | 1035.8 |
| 3.35 | 5695.7 | 1028.8 |
| 3.67 | 5678.5 | 1017.6 |
| 3.98 | 5657.7 | 1005.9 |
| 4.30 | 5559.5 | 1003.7 |
| 4.61 | 5459.4 | 1002.0 |
| 4.93 | 5364.5 | 1000.5 |
| 5.28 | 3355.9 | 883.5 |
| 5.63 | 3301.9 | 882.2 |
| 5.98 | 3248.8 | 881.0 |
| 6.33 | 3218.4 | 875.9 |
| 6.68 | 3193.1 | 870.5 |
| 7.03 | 3167.1 | 865.6 |
| 7.38 | 3114.3 | 865.7 |
| 7.73 | 3062.3 | 865.8 |
| 8.08 | 3019.3 | 864.4 |
| 8.43 | 3004.0 | 857.7 |
| 8.78 | 2989.8 | 851.0 |
| 9.13 | 2981.1 | 843.3 |
| 9.48 | 2985.8 | 833.8 |

**Table 6 (Cont.)**

**Main Steamline Break Inside Containment Mass and Energy Release Results for Limiting Case (1.4ft$^2$ Break at 0% Power)**

| Time (sec) | Break Flow (lbm/sec) | Break Enthalpy (Btu/lbm) |
|---|---|---|
| 9.83 | 2998.5 | 824.2 |
| 10.55 | 2886.8 | 827.2 |
| 14.22 | 2376.7 | 848.6 |
| 17.64 | 1972.2 | 881.4 |
| 21.10 | 1748.7 | 883.1 |
| 24.94 | 1567.5 | 882.9 |
| 29.48 | 1475.6 | 853.1 |
| 33.55 | 1398.6 | 831.3 |
| 36.76 | 1253.2 | 858.4 |
| 40.29 | 1217.0 | 837.3 |
| 43.17 | 1081.1 | 877.5 |
| 45.92 | 969.0 | 920.6 |
| 48.47 | 1020.0 | 862.8 |
| 50.97 | 1030.7 | 838.3 |
| 53.48 | 964.0 | 856.9 |
| 56.02 | 1020.6 | 808.0 |
| 59.76 | 935.1 | 833.3 |
| 67.27 | 726.6 | 943.6 |
| 74.79 | 669.3 | 963.3 |
| 82.30 | 580.4 | 1039.3 |
| 89.80 | 506.9 | 1127.6 |
| 97.30 | 460.6 | 1194.8 |
| 102.30 | 455.4 | 1194.7 |
| 112.30 | 443.7 | 1194.3 |
| 122.30 | 429.6 | 1193.8 |
| 132.30 | 415.9 | 1193.1 |
| 142.30 | 401.9 | 1192.5 |

**Table 6 (Cont.)**

**Main Steamline Break Inside Containment Mass and Energy Release Results for Limiting Case (1.4ft$^2$ Break at 0% Power)**

| Time (sec) | Break Flow (lbm/sec) | Break Enthalpy (Btu/lbm) |
|---|---|---|
| 152.30 | 387.8 | 1191.8 |
| 162.30 | 373.8 | 1191.1 |
| 172.30 | 360.0 | 1190.4 |
| 182.30 | 346.4 | 1189.8 |
| 192.30 | 333.2 | 1189.1 |
| 232.30 | 287.1 | 1186.7 |
| 332.30 | 212.3 | 1179.8 |
| 432.30 | 167.3 | 1175.7 |
| 532.30 | 139.8 | 1173.3 |
| 631.90 | 119.9 | 1171.5 |
| 730.90 | 66.8 | 1166.8 |
| 829.90 | 12.2 | 1166.2 |
| 928.10 | 54.0 | 1166.2 |
| 1024.66 | 12.2 | 1166.2 |
| 1120.88 | 12.1 | 1166.2 |
| 1200.00 | 12.1 | 1166.2 |

**Table 7**

**Peak Containment Pressures and Temperatures for MSLB Cases**

| Description | GOTHIC Peak Press (psig) | GOTHIC Peak Temp (°F) |
|---|---|---|
| 0.1 $ft^2$ break at 0% power | 18.33 | 214.3 |
| 0.5 $ft^2$ break at 0% power | 35.71 | 251.1 |
| 0.8 $ft^2$ break at 0% power | 43.06 | 263.0 |
| 1.1 $ft^2$ break at 0% power | 45.16 | 265.8 |
| 1.4 $ft^2$ break at 0% power | 45.68 | 266.6 |
| 1.4 $ft^2$ break at 30% power | 41.87 | 260.9 |
| 1.4 $ft^2$ break at 70% power | 41.93 | 260.9 |
| 1.4 $ft^2$ break at 102% power | 43.33 | 263.4 |

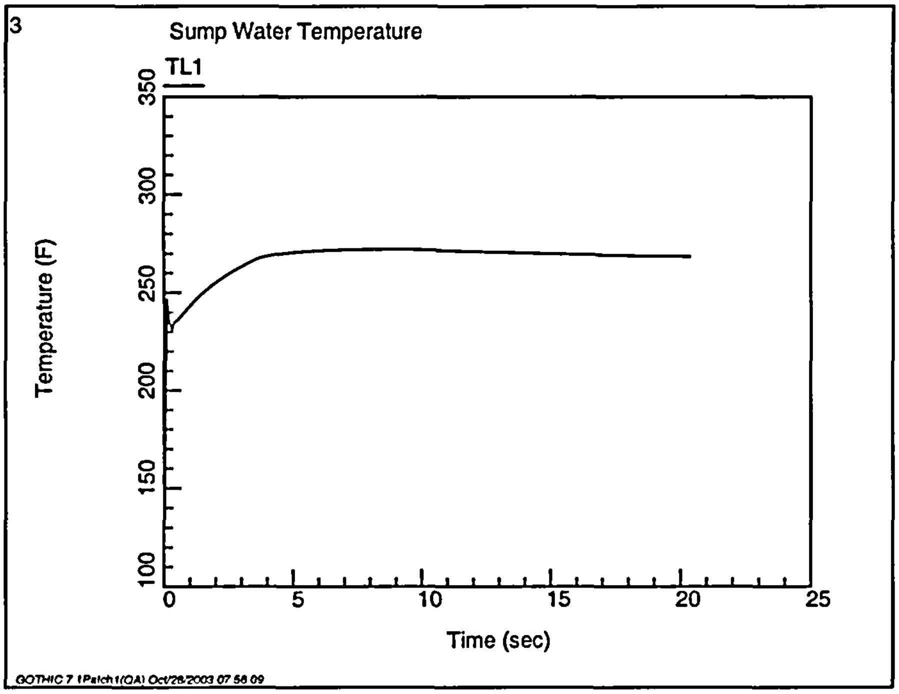


**Figure 6.4-3 DEHL Break Containment Sump Temperature**

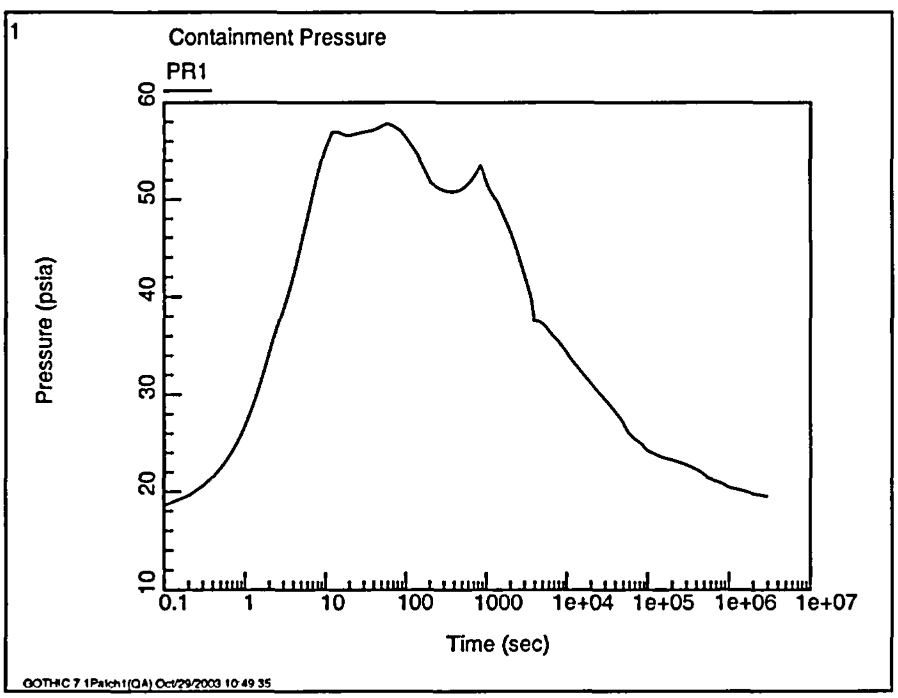


**Figure 6.4-4 DEPS Break with Minimum Safeguards-Containment Pressure**

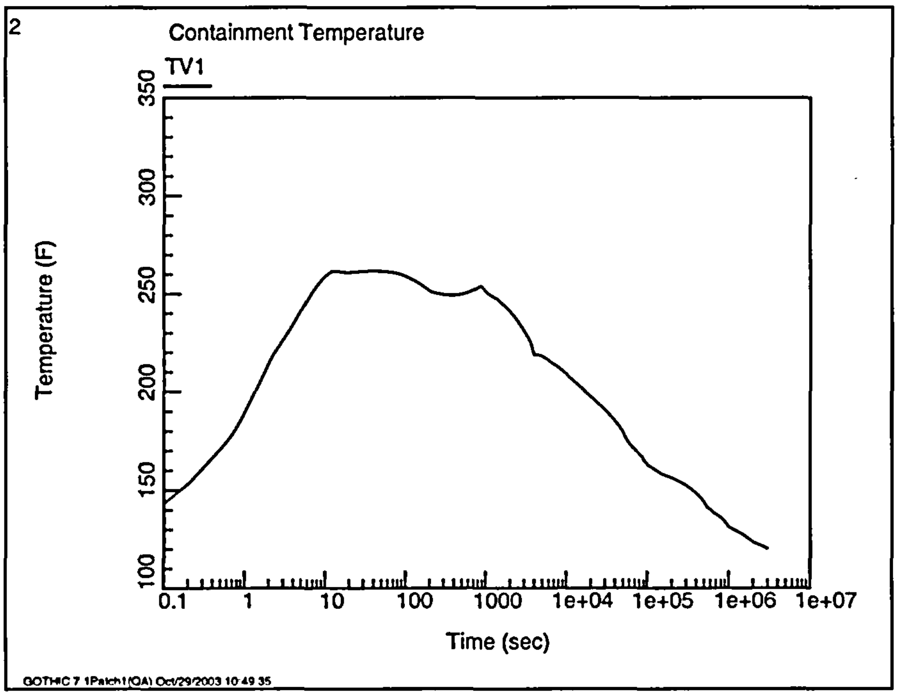


**Figure 6.4-5 DEPS Break with Minimum Safeguards-Containment Atmosphere Temperature**

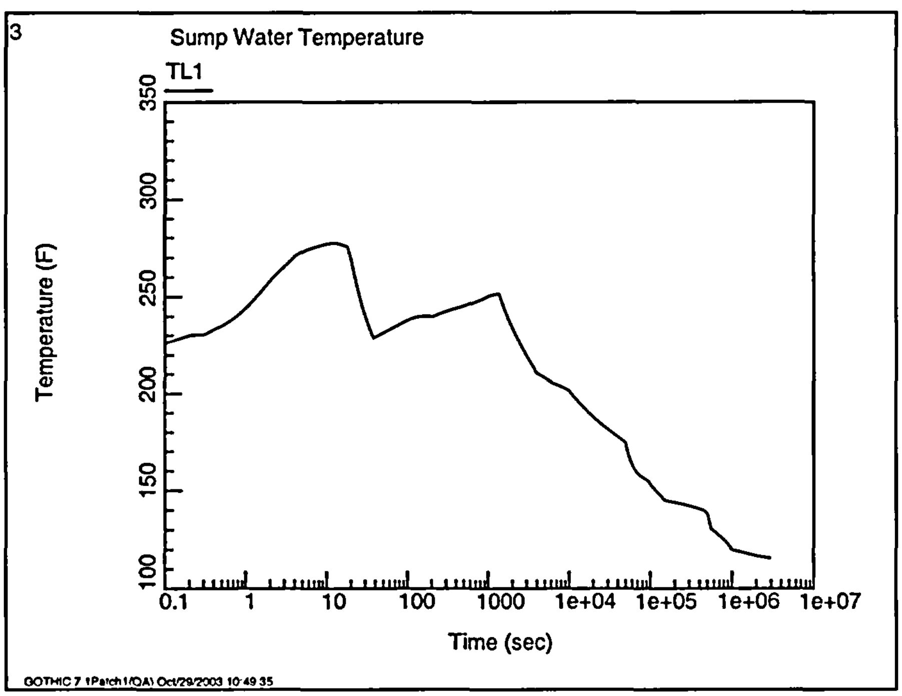


**Figure 6.4-6 DEPS Break with Minimum Safeguards-Containment Sump Pressure**

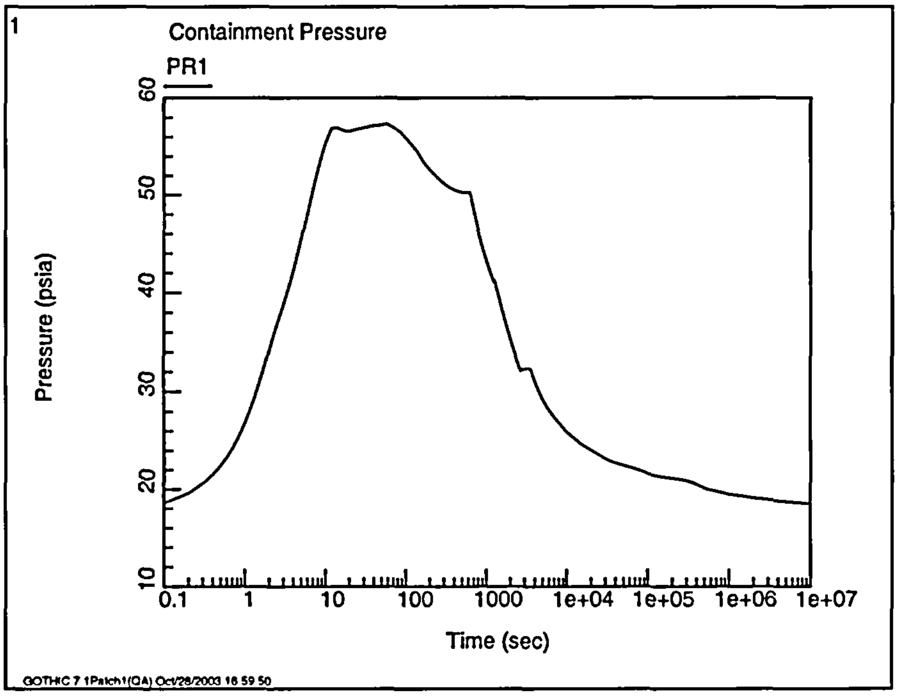


**Figure 6.4-7 DEPS Break with Maximum Containment Safeguards, 1 Spray Pump Failure Containment Pressure**

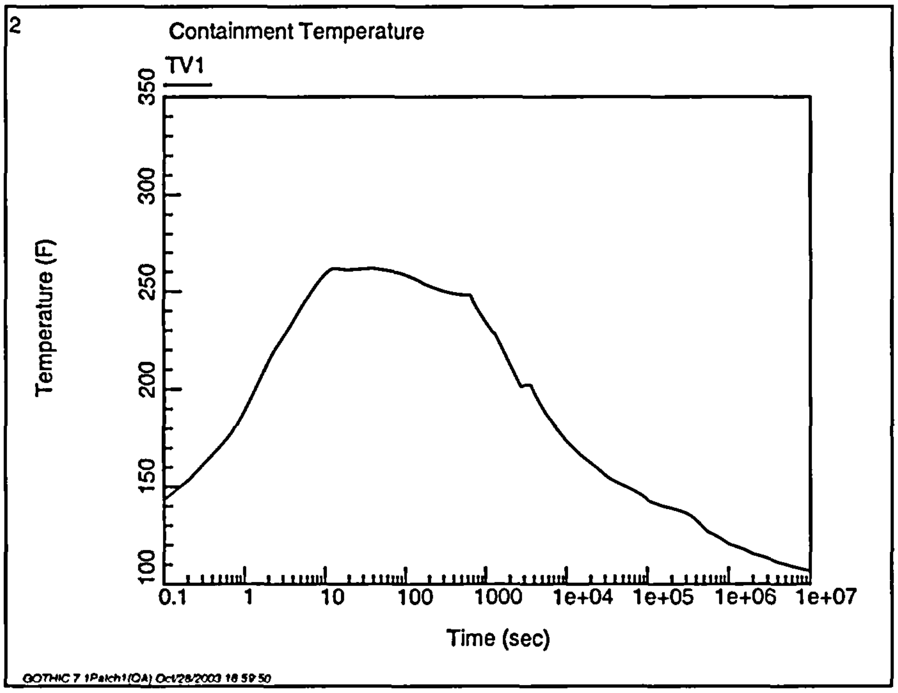


**Figure 6.4-8 DEPS Break with Maximum Containment Safeguards, 1 Spray Pump Failure Containment Atmosphere Temperature**

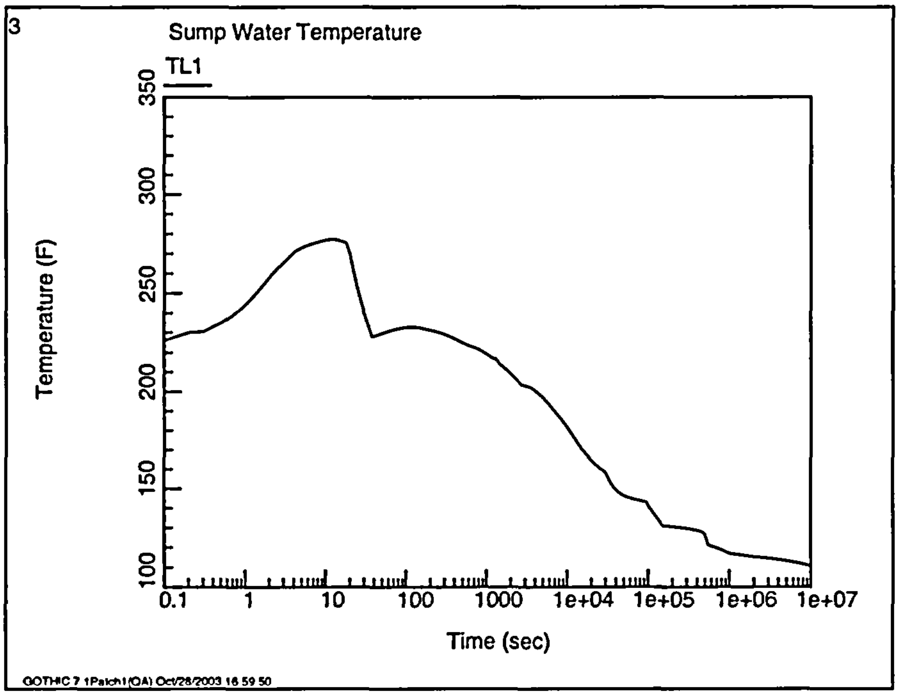


**Figure 6.4-9 DEPS Break with Maximum Containment Safeguards, 1 Spray Pump Failure Containment Sump Temperature**

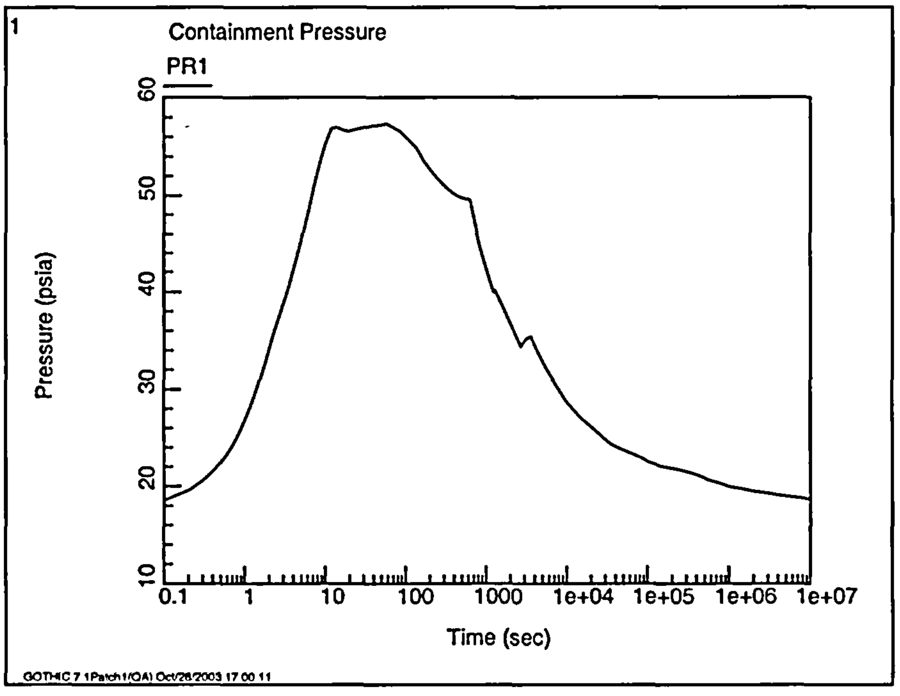


**Figure 6.4-10 DEPS Break with Maximum Containment Safeguards, 1 Fan Cooler Failure Containment Pressure**

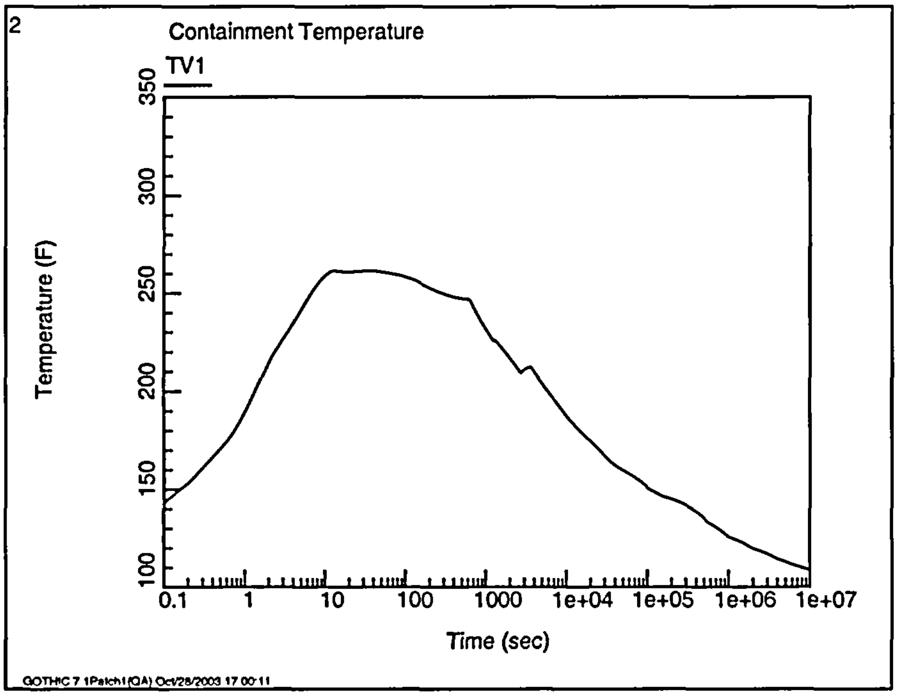


**Figure 6.4-11 DEPS Break with Maximum Containment Safeguards, 1 Fan Cooler Failure Containment Atmosphere Temperature**

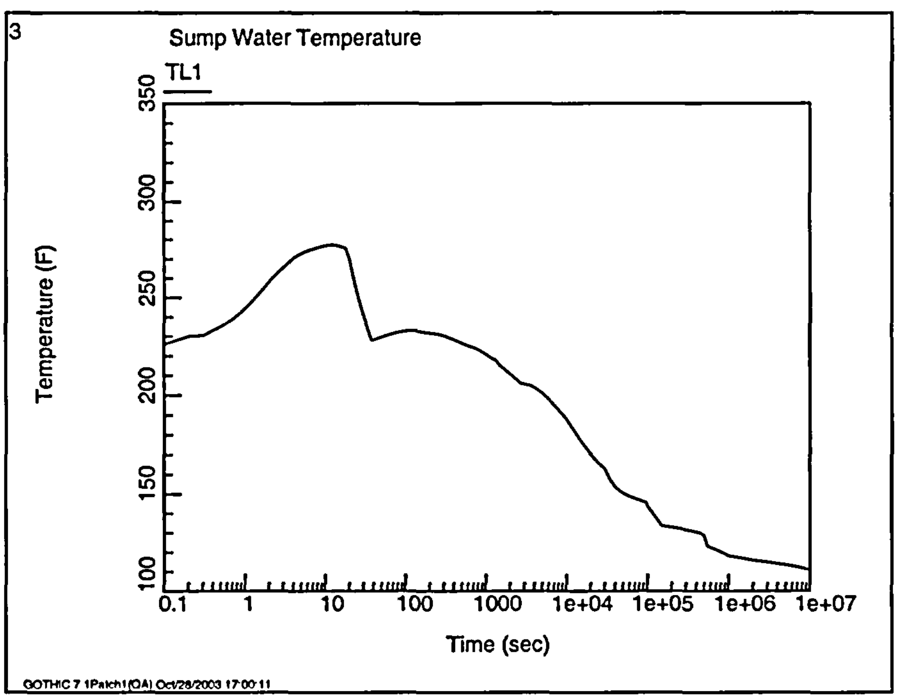


**Figure 6.4-12 DEPS Break with Maximum Containment Safeguards, 1 Fan Cooler Failure Containment Sump Temperature**

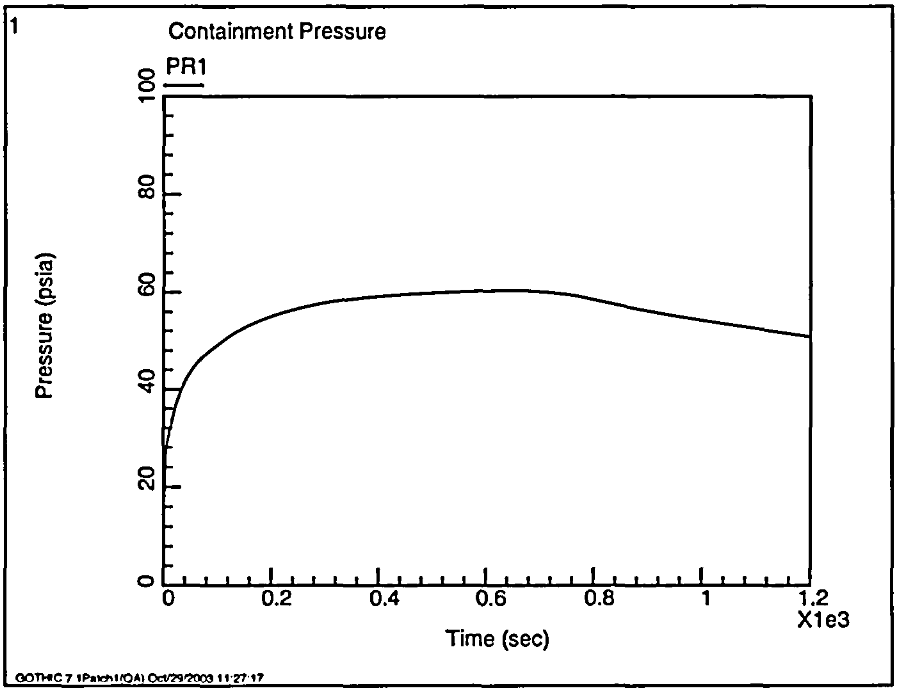


**Figure 6.4-13 Limiting MSLB Containment Pressure Response**

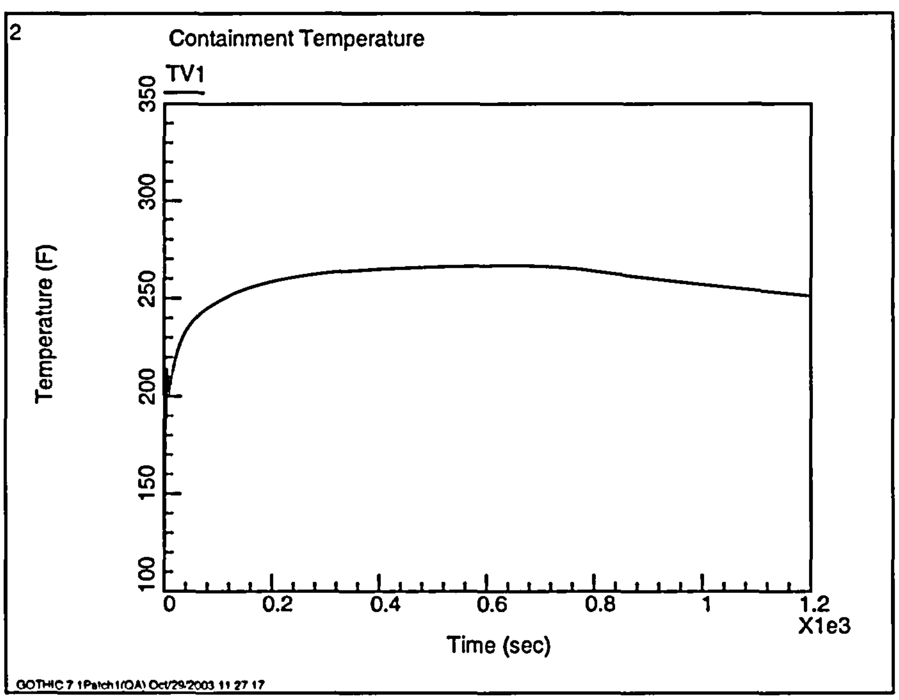


**Figure 6.4-14 Limiting MSLB Containment Temperature Response**

ATTACHMENT 5

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

**November 5, 2003**

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Responses to Requests for Additional Information and Supplemental Information Regarding LAR 195

Supplement to Loss of Normal Feedwater (LONF) and Loss of AC Power (LOAC) Safety Analyses

22 Pages to Follow

# Supplement to Loss of Normal Feedwater (LONF) and Loss of AC Power (LOAC) Safety Analyses

## Introduction and Background

The Kewaunee safety analyses for the Loss of Normal Feedwater transient (LONF, USAR Section14.1.10) and the Loss of AC Power to the Plant Auxiliaries transient (LOAC, USAR Section14.1.12) were revised after it was discovered that a least negative Doppler-only power coefficient (DPC) expression was inadvertently assumed, rather than the most negative DPC expression. The DPC is expressed in terms of reactivity change per percent power change, and is more negative at beginning of cycle life (BOL) conditions. In the LONF and LOAC analyses, the nuclear power reduction following reactor trip causes a positive reactivity insertion via the DPC feedback, which tends to reduce the effective trip reactivity worth. The more negative the DPC, the greater the positive reactivity insertion when power decreases, and the lower the effective trip reactivity worth. A lower effective trip reactivity worth leads to a longer time necessary to reduce the neutron population. The end result is greater power input in the seconds that follow the reactor trip before only decay heat remains as a source of power generation in the core. As the integral power generation is critical in the LONF and LOAC analyses, the assumption of a most negative DPC is more conservative.

The LONF and LOAC safety analyses presented in this supplement report supersede the previously reported safety analyses for these transients provided in reference 1, appendix 6A (LONF) and reference 2 attachment B, section 5.1.11 (LOAC). The results from the revised safety analyses for the LONF and LOAC transients continue to satisfy the applicable safety analysis acceptance criteria. No changes to the proposed Stretch power uprate technical specification changes and to the existing stretch power uprate significant hazards determination (reference 1) are required as a result of these revised safety analyses.

The presentation of the revised safety analyses in this supplement report uses the KNPP USAR format with the corresponding USAR section numbering (e.g. LONF and LOAC, documented in sections 14.1.10, 14.1.12 of the KNPP USAR, are also documented with that format and numbering in this report). These revised safety analyses will be incorporated into the stretch power uprate KNPP USAR update following implementation of the stretch power uprate.

## 14.1.10 Loss of Normal Feedwater

**Accident Description:**

A loss of normal feedwater (from a pipe break, pump failure, or valve malfunction) results in a reduction of the ability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, core damage could possibly occur from a sudden loss of heat sink. If an alternate supply of feedwater were not supplied to the steam generators, residual heat following reactor trip and reactor coolant pump (RCP) heat would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the Reactor Coolant System (RCS). A significant loss of water from the RCS could conceivably lead to core damage. Controlled shutdown of the reactor and RCS stabilization are also very challenging with a water-solid pressurizer. Since the reactor is tripped well before the steam generator heat transfer capability is reduced, the primary system never approaches a condition where the departure from nucleate boiling ratio (DNBR) limit may be violated.

The following features provide the necessary protection against a loss of normal feedwater:

1. Reactor trip on lo-lo water level in two-out-of-three level channels in either steam generator.

2. Reactor trip on steam flow-feedwater flow mismatch coincident with low water level in either steam generator.

3. Two motor-driven auxiliary feedwater (AFW) pumps, which are started on:

   a. Lo-lo level in two-out-of-three level channels in either steam generator
   b. Opening of both feedwater pump circuit breakers
   c. Any safety injection signal
   d. Loss of offsite power
   e. Steam generator anticipated transient without scram (ATWS) mitigation system actuation circuitry (AMSAC) lo-lo water level
   f. Manual actuation

4. One turbine-driven AFW pump, which is started on:

   a. Lo-lo level in two-out-of-three channels in both steam generators
   b. Loss of voltage on both 4 kV buses
   c. Steam generator AMSAC lo-lo water level
   d. Manual actuation

The Auxiliary Feedwater (AFW) system is started automatically on the signals described above. Below 15 percent of rated thermal power (RTP), select AFW valves (AFW-2A and AFW-2B, and AFW-10A and AFW-10B) can be placed in the closed position (per Technical Specification 3.4.b), thereby precluding automatic delivery of AFW flow to the steam generators. Also below 15 percent of RTP, the control switches for the AFW pumps can be placed in the "pull out" position to prevent filling the steam generators. For these conditions below 15 percent of RTP, operator action to manually establish AFW flow from at least two AFW pumps within 800 seconds (13.3 minutes) after a reactor trip has been determined to be acceptable based on the 100 percent of RTP loss-of-normal feedwater analysis, in which an 800-second AFW delay has been assumed.

Following a loss of offsite power, the emergency diesel generators supply electrical power to the two motor-driven AFW pumps. The turbine-driven AFW pump is powered via steam flow from the secondary system that exhausts to the atmosphere. All of the AFW pumps are normally aligned to take suction from the condensate storage tank (CST) for delivery to the steam generators.

The analysis shows that following a loss of normal feedwater, the AFW system is capable of removing the stored energy, residual decay heat and RCP heat following reactor trip. The pressurizer is prevented from becoming water-solid, which could lead to overpressurization of the RCS and a subsequent loss of water from the RCS via a failed-open pressurizer pressure relief or safety valve.

**Method of Analysis:**

The loss-of-normal-feedwater transient is analyzed using the RETRAN computer code. The RETRAN model simulates the RCS, neutron kinetics, pressurizer, pressurizer relief and safety valves, pressurizer heaters, pressurizer spray, steam generators, feedwater system and main steam safety valves (MSSVs). The code computes pertinent plant variables including steam generator mass, pressurizer water volume and reactor coolant average temperature.

The major assumptions are summarized below.

1. The plant is initially operating at 100.6 percent of 1780 MWt NSSS (includes 10 MWt of RCP heat)

2. Reactor trip occurs on steam generator lo-lo water level at 0 percent of narrow range span (NRS). Turbine trip occurs coincident with reactor trip.

3. A conservative core residual heat generation is assumed, based on the ANS 5.1-1979 decay heat model plus 2 sigma.

4. AFW flow from two motor-driven AFW pumps is initiated with flow split equally between the two steam generators (equal split is the limiting case) 800 seconds after the reactor trip on lo-lo steam generator water level. This AFW flow assumption accounts for the limiting single failure that is the loss of the turbine-driven AFW pump. The AFW is modeled as a function of steam generator pressure, and the flow with the first (lowest setpoint) MSSVs open is approximately 170 gpm. The AFW enthalpy is assumed to be 90.8 BTU/lbm (120°F and 1100 psia)

5. Secondary system steam relief is achieved through the MSSVs. The MSSVs model includes a + 1 percent setpoint tolerance and a 5 psi ramp for the valve to open. Steam relief through the steam generator power-operated relief valves (PORVs) or condenser dump valves is assumed to be unavailable.

6. The initial reactor coolant average temperature is assumed to be 6°F higher than the nominal full power value of 573.0°F because this results in a greater expansion of the RCS water during the transient, thus resulting in a higher pressurizer water level.

7. The initial pressurizer pressure is assumed to be 50 psi above the nominal value of 2250 psia. A sensitivity study was performed that demonstrated that a high initial pressurizer pressure is conservative. An additional 0.1-psi uncertainty has been determined to be negligible.

8. The initial pressurizer water level is assumed to be 5 percent of span above the nominal value of 48 percent of span, which corresponds to the high nominal full-power vessel average temperature of 573°F. A high initial pressurizer water level is conservative because it minimizes the initial margin to filling the pressurizer water-solid.

9. Normal reactor control systems are not assumed to function. However, the pressurizer PORVs, pressurizer heaters and pressurizer sprays are assumed to operate normally. This assumption results in a conservative transient with respect to the peak pressurizer water level. If these control systems did not operate, the pressurizer safety valves would maintain peak RCS pressure around the actuation setpoint throughout the transient, which would limit the peak pressurizer water volume.

10. The initial steam generator water level is assumed to be 7 percent of narrow range span (NRS) above the nominal value of 44 percent of NRS. A high initial steam generator water level is conservative because it maximizes the time to reach the steam generator lo-lo water level, thereby maximizing the RCS heatup.

11. Reactor trip on steam flow-feedwater flow mismatch coincident with low water level in either steam generator is not credited.

The loss-of-normal-feedwater analysis is performed to demonstrate the adequacy of the Reactor Protection System to trip the reactor and the Engineered Safeguards Features Actuation System (AFW system) to remove long-term decay heat, stored energy and RCP heat following reactor trip. The actuation of the AFW system prevents excessive heatup or overpressurization of the RCS. As such, the assumptions used in the analysis are designed to maximize the time to reactor trip and to minimize the energy removal capability of the AFW system. These assumptions maximize the possibility of water relief from the RCS by maximizing the expansion of the RCS inventory, as noted in the assumptions listed above.

**Results:**

Figures 14.1.10-1 through 14.1.10-6 show the significant plant responses following a loss of normal feedwater. The calculated sequence of events is listed in Table 14.1.10-1.

Following the reactor and turbine trip from full load, the water level in each steam generator falls due to the reduction of the steam generator void fraction in the tube bundle region, and because the steam release through the MSSVs, which open to dissipate the RCS stored and generated heat. Eight hundred seconds after the initiation of the lo-lo steam generator water level reactor trip, flow from the two motor-driven AFW pumps is credited, thus reducing the rate of water level decrease in the steam generators.

The capacity of two motor-driven AFW pumps is sufficient to dissipate core residual heat, stored energy and RCP heat without water relief through the pressurizer PORVs or safety valves. Figure 14.1.10-4 shows that at no time is there water relief from the pressurizer, as the peak pressurizer water volume is less than the limit of 1010.1 $ft^3$. Plant emergency operating procedures may be followed to further cool down the plant. The peak Main Steam System (MSS) pressure is less than 110 percent of the steam generator design pressure. Also, the analysis shows that the RCS overpressurization limit is not challenged during this transient. However, note that the pressurizer sprays and PORVs are assumed to be operable so as to maximize the potential for pressurizer filling. This event is bounded by the loss of external electrical load with respect to peak RCS and MSS pressures.

## Conclusions:

The results of the loss-of-normal feedwater analysis show that all applicable acceptance criteria are satisfied. The AFW capacity is sufficient to dissipate core residual heat, stored energy and reactor coolant pump heat such that reactor coolant water is not relieved through the pressurizer relief or safety valves.

**Table 14.1.10-1**

**Sequence of Events**

**Loss of Normal Feedwater**

| Event | Time (seconds) |
|---|---|
| Main feedwater flow stops | 20 |
| Lo-Lo steam generator water level trip setpoint reached | 53.0 |
| Rods begin to drop | 54.5 |
| Two steam generators begin to receive auxiliary feedwater | 854.5 |
| Peak water level in the pressurizer occurs | 1157.5 |
| Core heat decreases to auxiliary feedwater heat removal capacity | ~1300 |

Loss of Normal Feedwater
Nuclear Power vs. Time

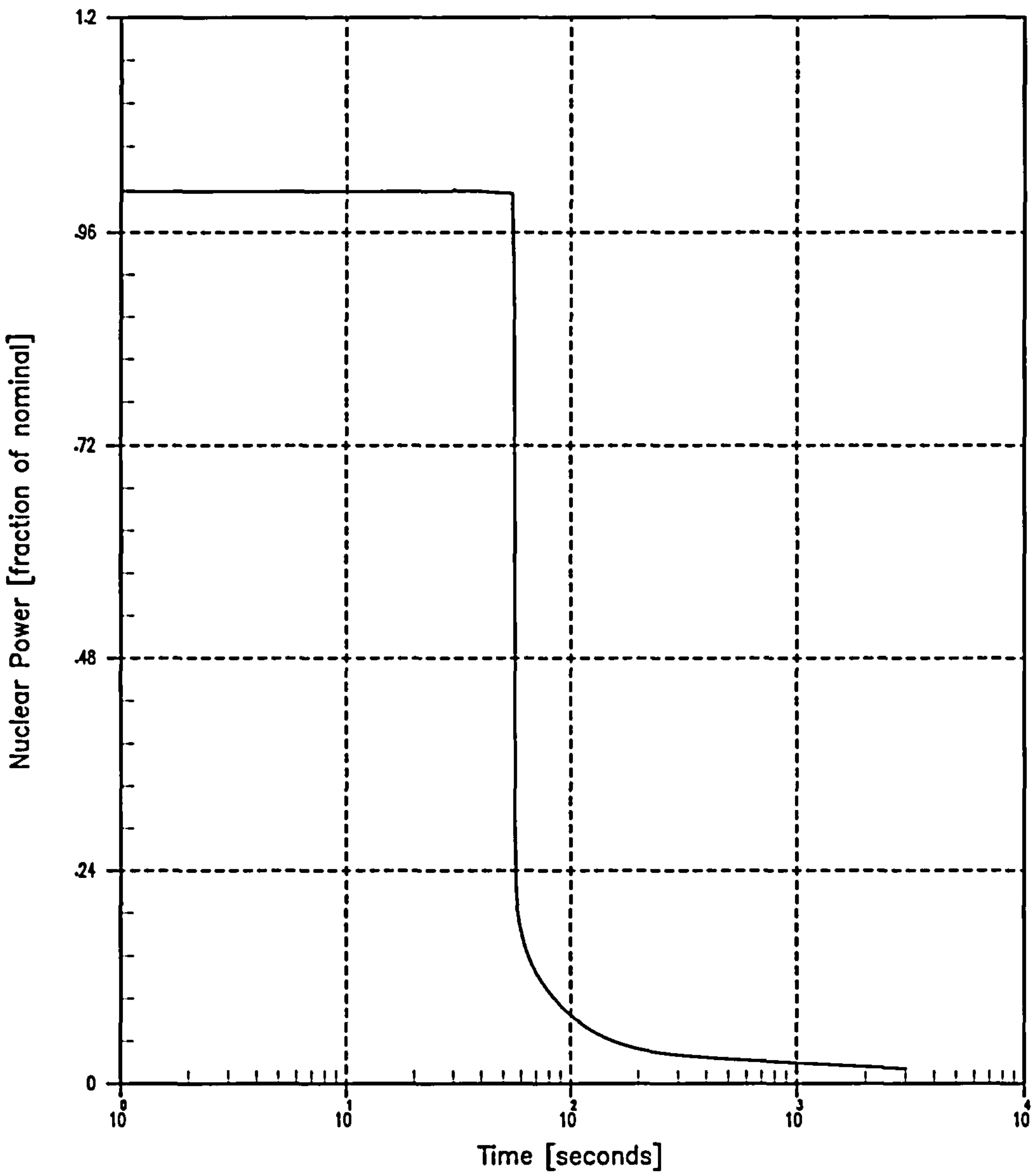


**Figure 14.1.10-1**

Loss of Normal Feedwater
Vessel Average Temperature vs. Time

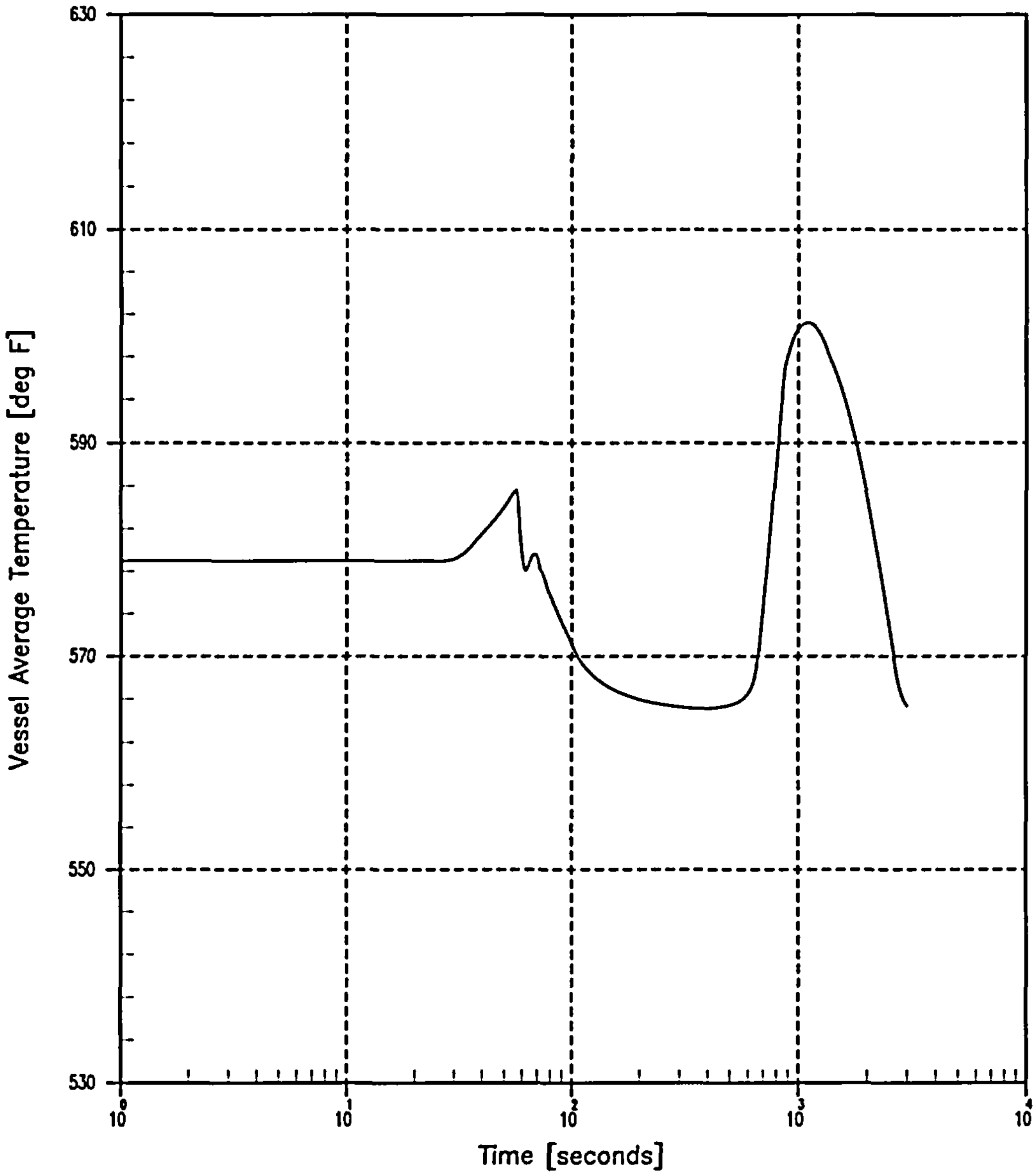


**Figure 14.1.10-2**

Loss of Normal Feedwater
Pressurizer Pressure vs. Time

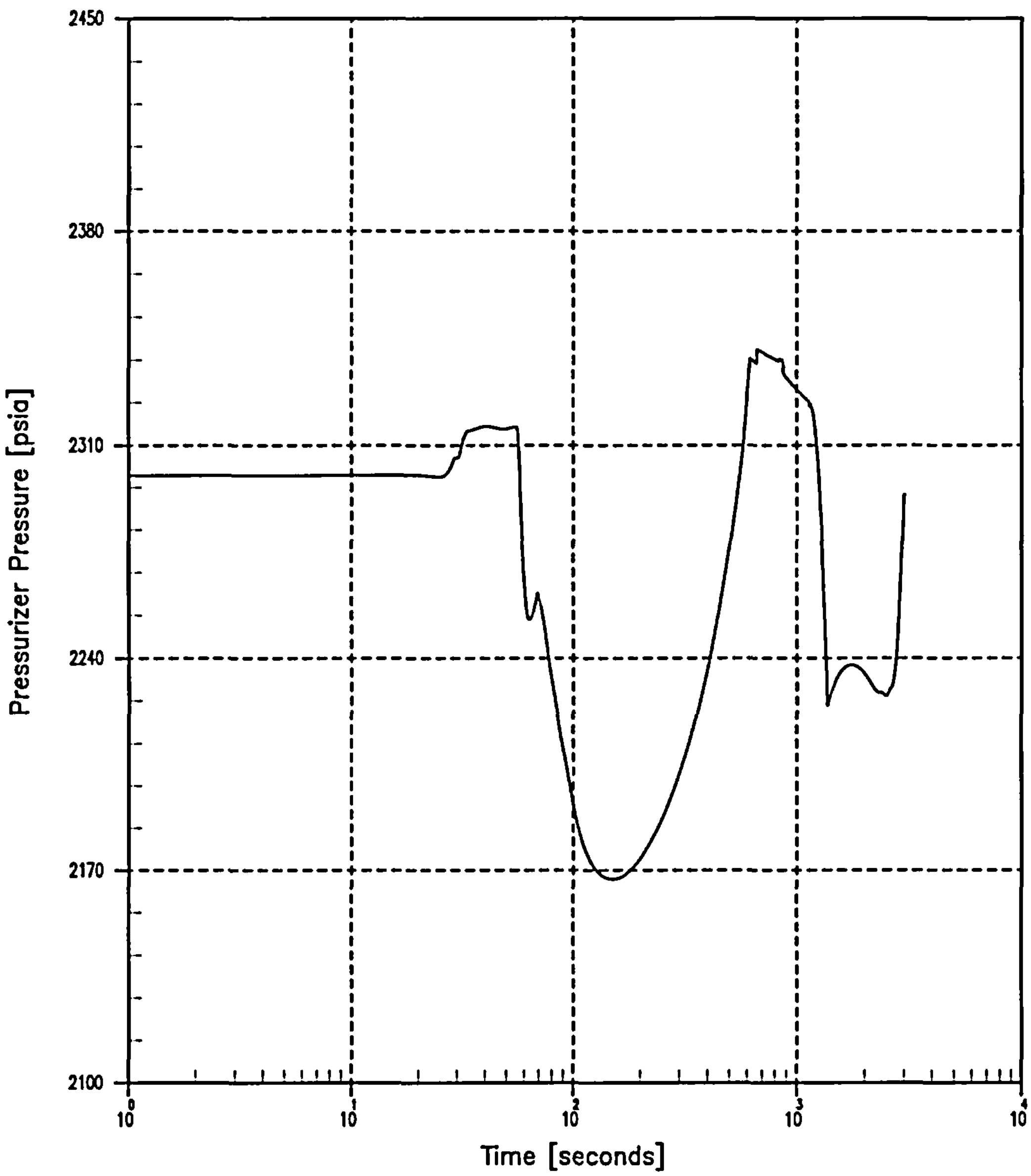


**Figure 14.1.10-3**

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Loss of Normal Feedwater
Pressurizer Water Volume vs. Time

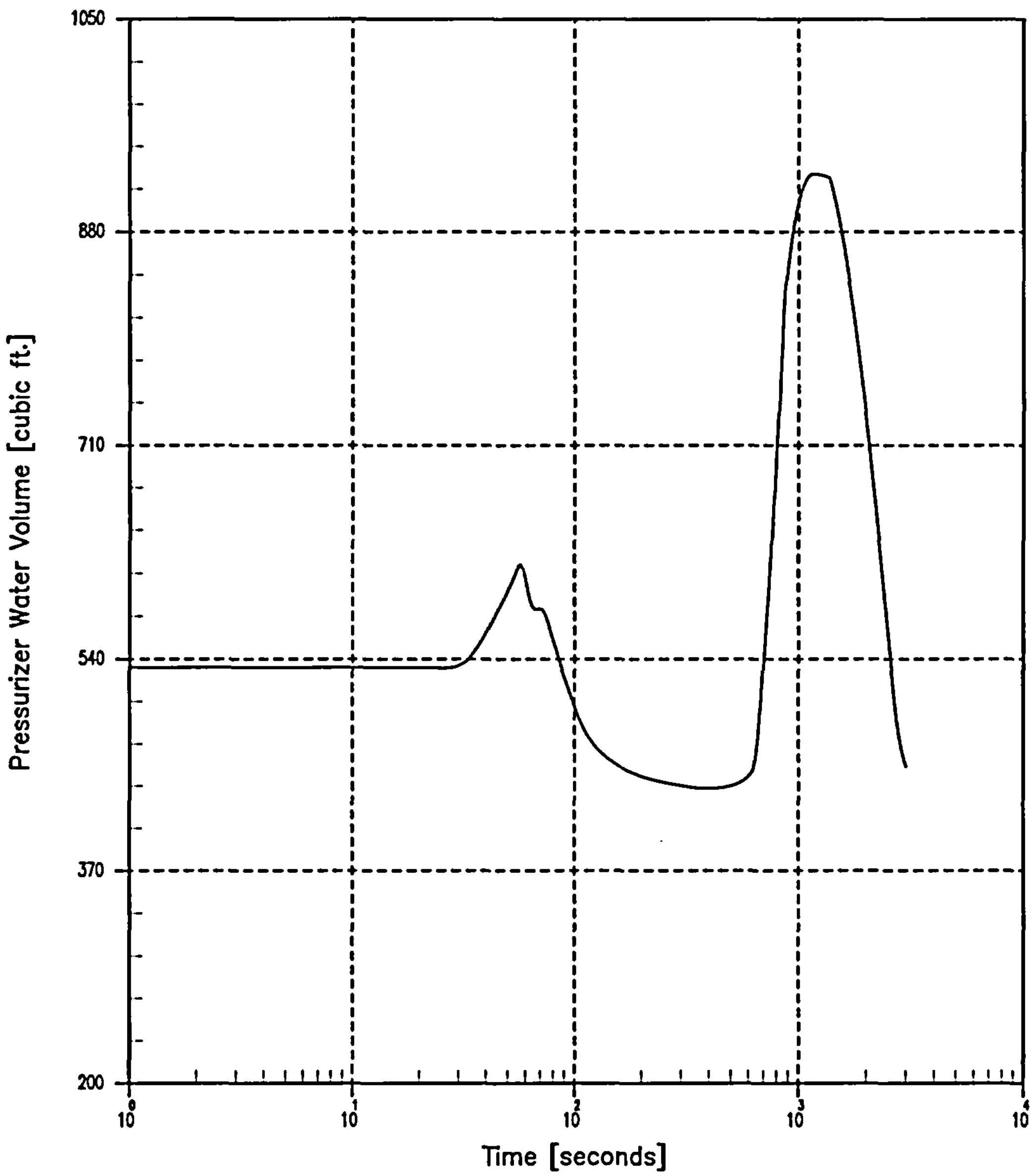


**Figure 14.1.10-4**

Loss of Normal Feedwater
Steam Generator Pressure vs. Time

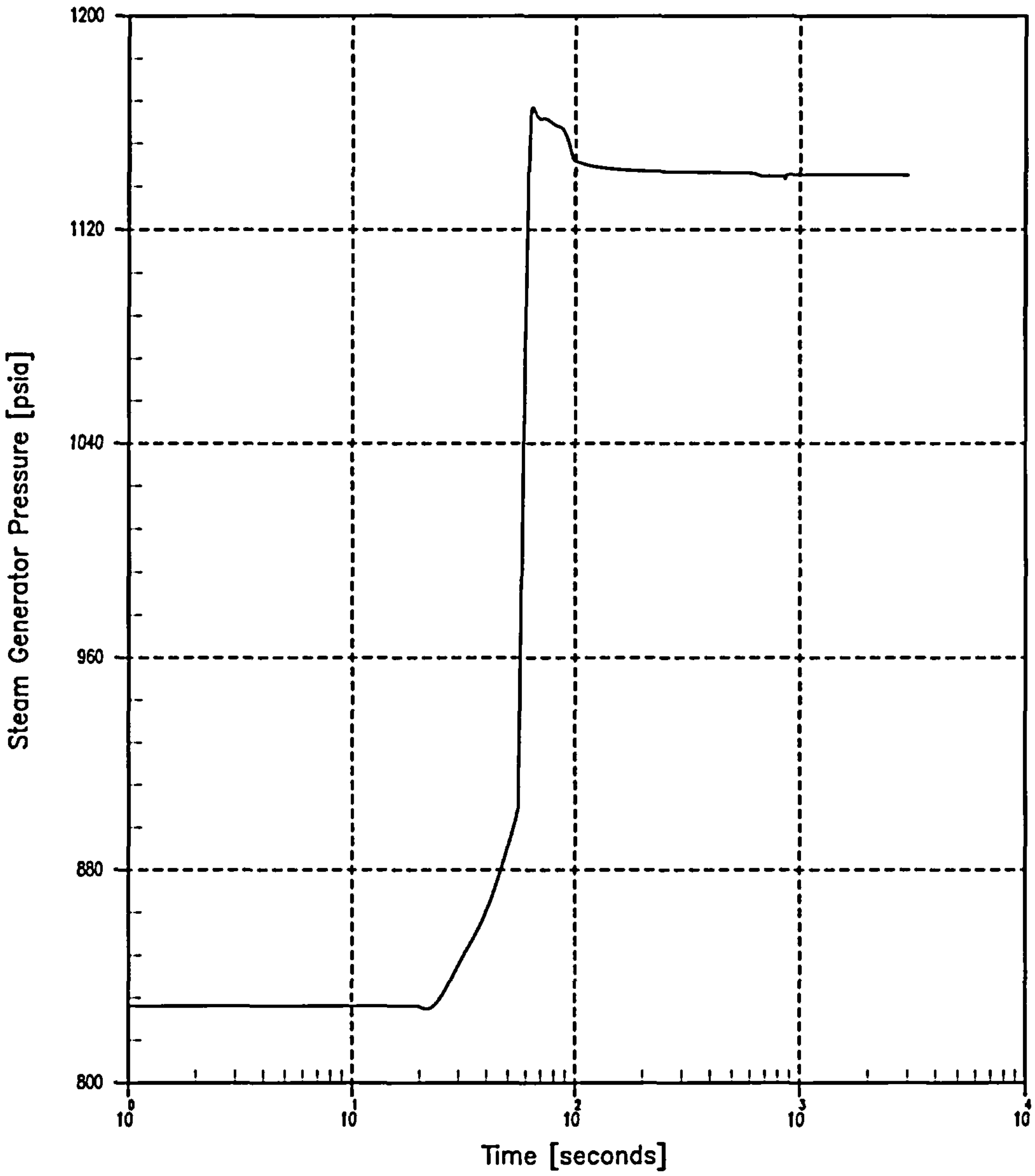


**Figure 14.1.10-5**

Loss of Normal Feedwater
Steam Generator Mass vs. Time

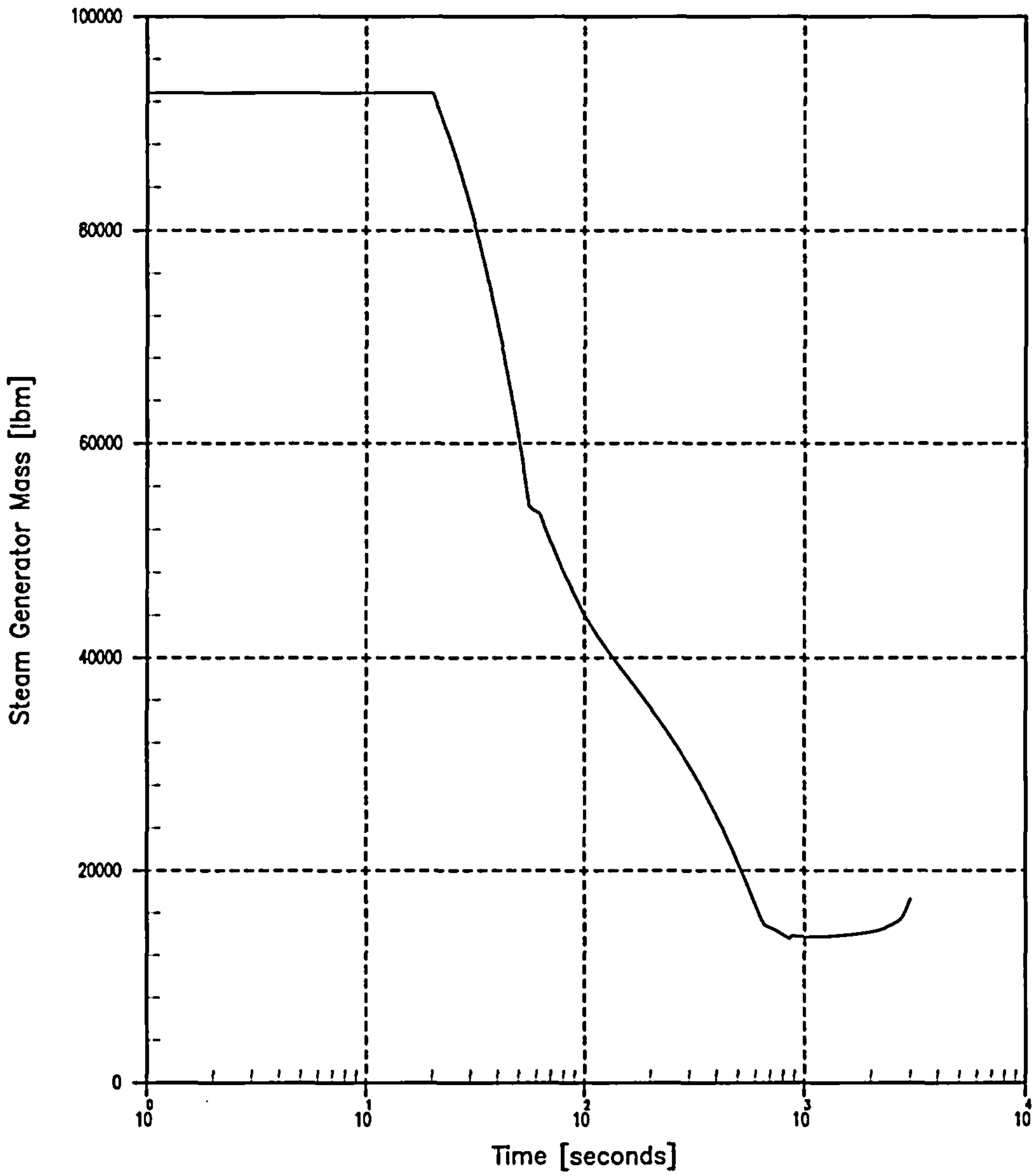


**Figure 14.1.10-6**

**14.1.12 Loss of All AC Power to the Plant Auxiliaries**

**Accident Description:**

A complete loss of non-emergency AC power results in the loss of all power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of the onsite AC distribution system.

The events following a loss of AC power with turbine and reactor trip are described in the sequence listed below.

1. Plant vital instruments are supplied from emergency power sources.

2. Steam dump to the condenser and steam generator power operated relief valves (PORVs) are unavailable; therefore, the main steam safety valves lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat.

3. As the no-load temperature is approached, the steam generator PORVs (or the safety valves, if the PORVs are not available) are used to dissipate the residual decay heat and maintain the plant at the hot shutdown condition.

4. The standby diesel generators, started on loss of voltage on the plant emergency busses, begin to supply plant vital loads.

The Auxiliary Feedwater (AFW) System is started automatically, as discussed in the loss of normal feedwater analysis (Section 14.1.10). The turbine-driven AFW pump utilizes steam from the secondary system and exhausts to the atmosphere. The motor-driven AFW pumps are supplied by power from the diesel generators. The pumps take suction directly from the condensate storage tank for delivery to the steam generators.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops. Following the reactor coolant pump (RCP) coastdown caused by the loss of AC power, the natural circulation capability of the Reactor Coolant System (RCS) removes residual and decay heat from the core, aided by the AFW in the secondary system.

**Method of Analysis:**

The loss of all AC power to the station auxiliaries transient is analyzed using the RETRAN computer code. The code simulates the RCS, neutron kinetics, pressurizer, pressurizer relief and safety valves, pressurizer heaters, pressurizer spray, steam generators, feedwater system and main steam safety valves. The code computes pertinent plant variables including steam generator mass, pressurizer water volume and reactor coolant average temperature.

Major assumptions made in the loss of all auxiliary AC power analysis are summarized below. :

1. The plant is initially operating at 102 percent of the 1780 MWt.

2. Reactor trip occurs on steam generator lo-lo level at 0 percent of narrow range span. Turbine trip occurs coincident with reactor trip.

3. A conservative core residual heat generation based on ANS 5.1-1979 decay heat plus 2 sigma is assumed.

4. The amount of heat transfer assumed to occur in the steam generators following the reactor coolant pump coastdown is based on RCS natural circulation conditions.

5. One minute after the lo-lo steam generator water level setpoint is reached, the AFW system provides 176 gpm of flow split equally between the two steam generators (equal split is the limiting case). The AFW flow assumption is conservative with respect to the worst case scenario for available AFW flow during a loss of all auxiliary AC power event, as the turbine-driven AFW pump (single failure) and the second motor-driven AFW pump are assumed to be unavailable. The AFW enthalpy is assumed to be 90.8 BTU/lbm (120°F and 1100 psia).

6. Secondary system steam relief is achieved through the main steam safety valves, which include a +2 percent setpoint tolerance, a 5 psi ramp for the valve to pop open and a pressure difference from the steam generator to the safety valves of approximately 42 psi. Steam relief through the steam generator power-operated relief valves (PORVs) or condenser dump valves is assumed unavailable.

7. The initial reactor coolant average temperature is assumed to be 6°F lower than the nominal value of 573.0°F because this results in a greater expansion of the RCS water during the transient, thus, resulting in a higher pressurizer water level.

8. The initial pressurizer pressure is assumed to be 50 psi above its nominal value.

9. Nominal reactor control systems are not assumed to function. However, the pressurizer PORVs, pressurizer heaters and pressurizer spray are assumed to operate normally. This assumption results in a conservative transient with respect to the peak pressurizer water level. If these control systems did not operate, the pressurizer safety valves would maintain peak RCS pressure around the actuation setpoint throughout the transient.

The assumptions used in the analysis are similar to the loss of normal feedwater (Section 14.1.10) except that power is assumed to be lost to the reactor coolant pumps at the time of reactor trip.

**Results:**

Figures 14.1.12-1 through 14.1.12-6 show the significant plant responses following a loss of all AC power to the station auxiliaries event.

The first few seconds after the loss of power to the reactor coolant pumps will closely resemble the simulation of the complete loss of flow accident (FSAR Section 14.1.8), where core damage due to rapidly increasing core temperature is prevented by promptly tripping the reactor.

After the reactor trip, stored and residual decay heat must be removed to prevent damage to either the RCS or the core. The peak pressurizer water volume is less than the limit of 1010.1 $ft^3$. The maximum steam generator pressure calculated was less than 110 percent of the design pressure of 1085 psig. The RCS overpressurization limit is not challenged during this transient.

The RETRAN code results show that the reactor coolant natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

**Conclusions:**

The results of the analysis show that a loss of all AC power to the station auxiliaries does not adversely affect the core, the RCS or the MSS. The AFW capacity is sufficient to dissipate core residual heat. Consequently, reactor coolant is not relieved through the pressurizer relief or safety valves.

**References:**

1) KNPP Letter #NRC-03-057 of 5/22/03, LAR 195 for Stretch Power Uprate (LAR 195)
2) Letter from Thomas Coutu (NMC) to Document Control Desk (NRC), "License Amendment Request 187a to the Kewaunee Nuclear Power Plant Technical Specifications, Conforming Technical Specification Changes for Use of Westinghouse Vantage + Fuel," dated February 27, 2003

Loss of AC Power to the Plant Auxiliaries
Nuclear Power vs. Time

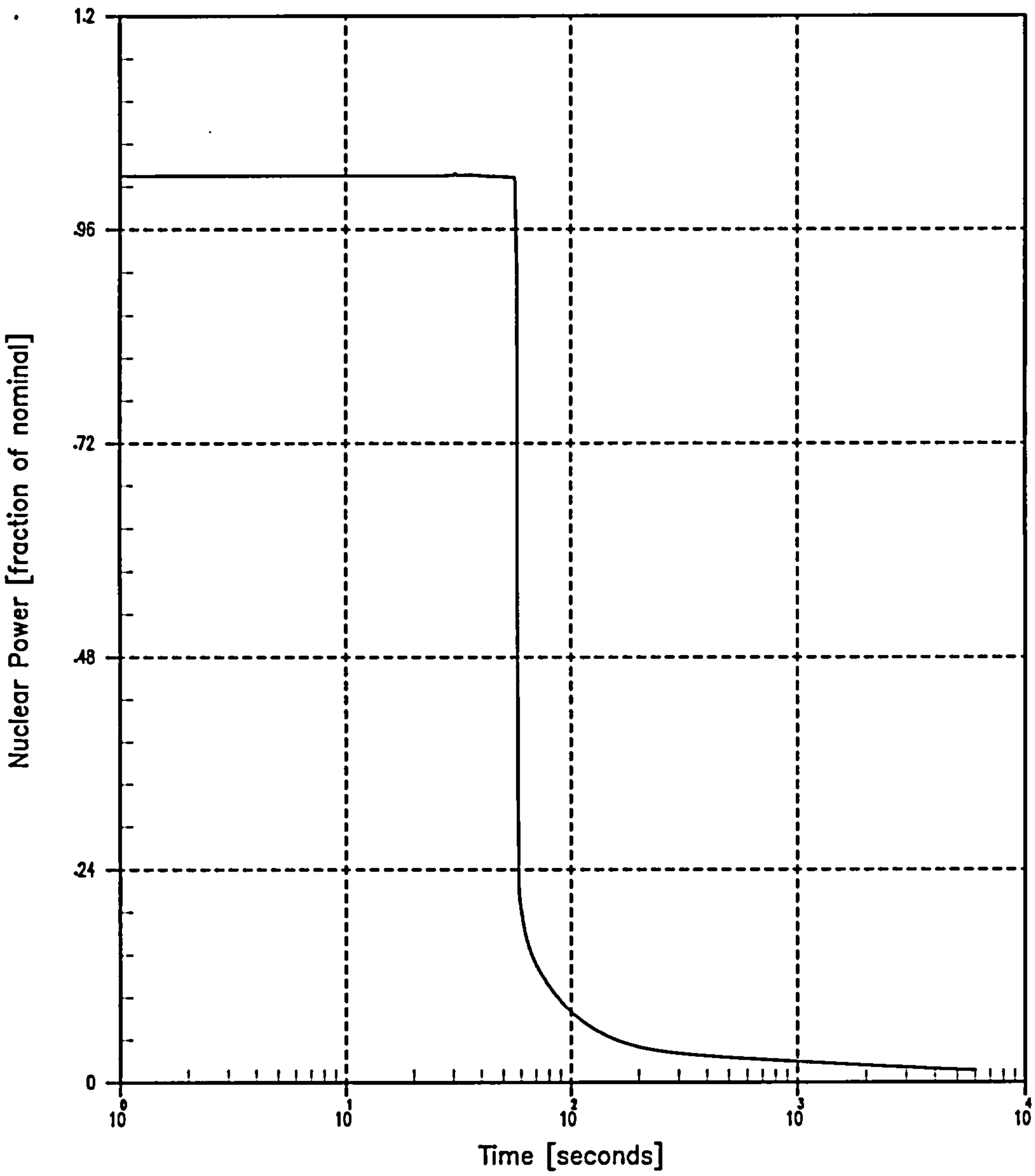


**Figure 14.1.12-1**

Loss of AC Power to the Plant Auxiliaries
Vessel Average Temperature vs. Time

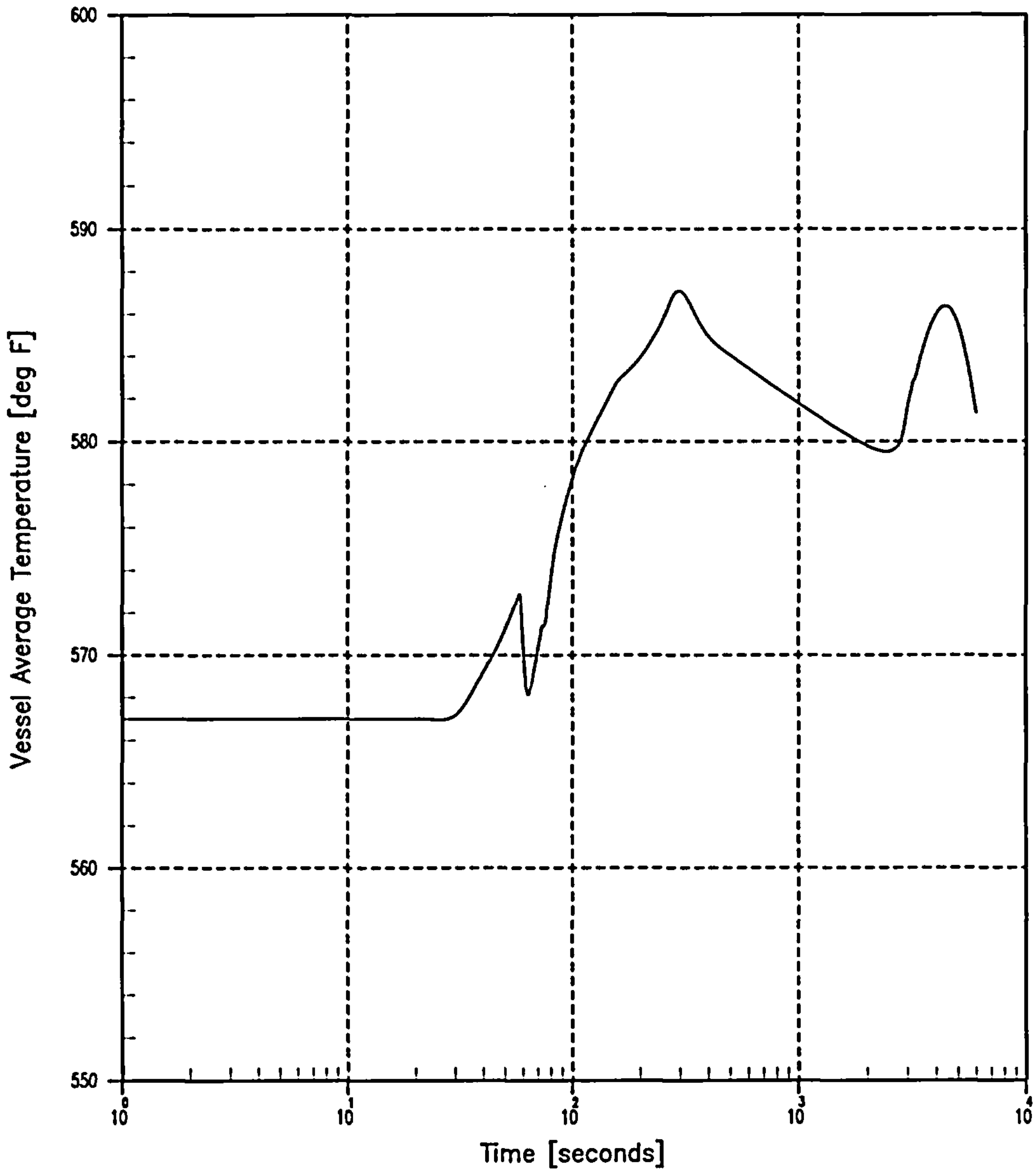


**Figure 14.1.12-2**

Loss of AC Power to the Plant Auxiliaries
Pressurizer Pressure vs. Time

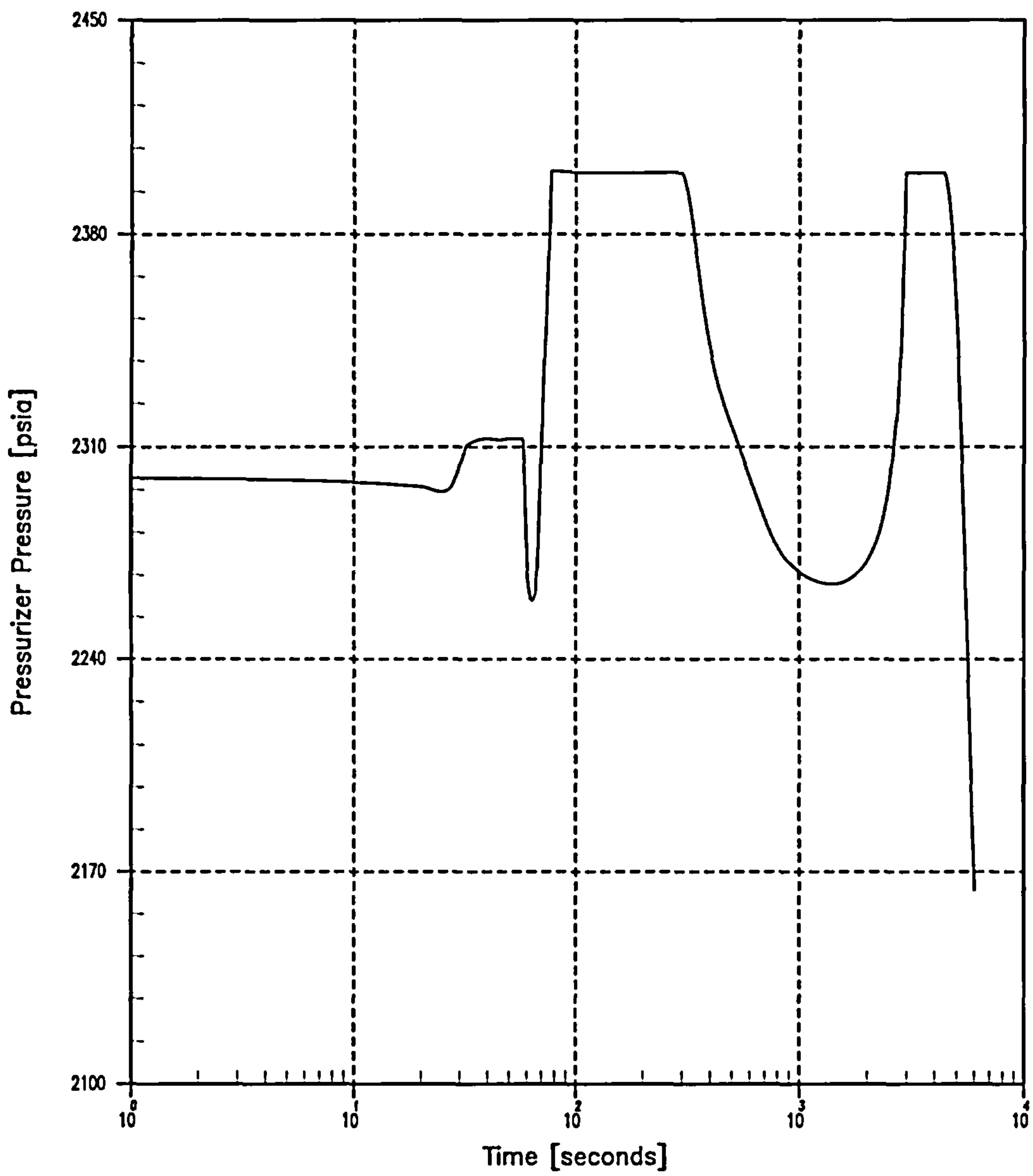


**Figure 14.1.12-3**

Loss of AC Power to the Plant Auxiliaries
Pressurizer Water Volume vs. Time

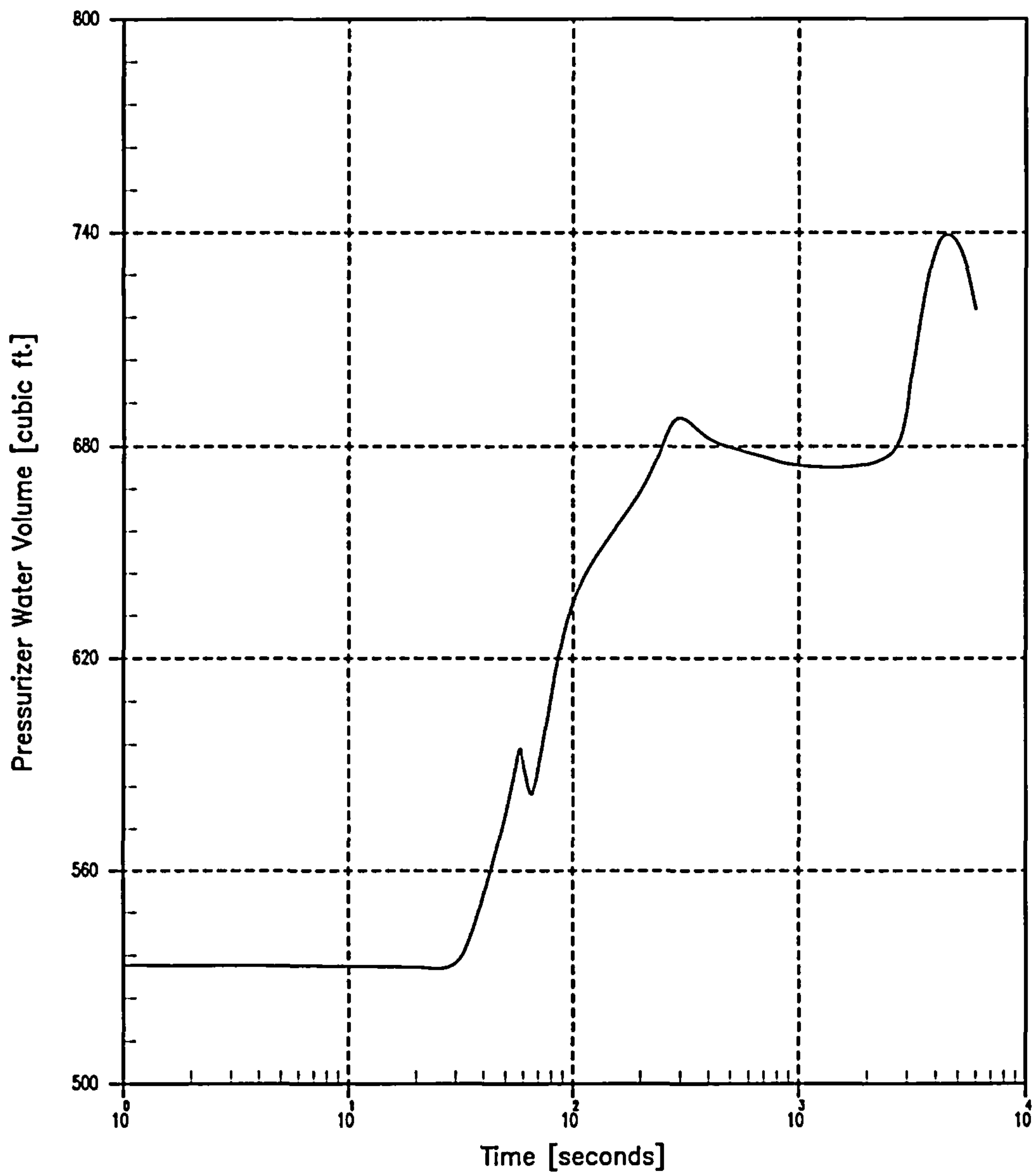


**Figure 14.1.12-4**

Loss of AC Power to the Plant Auxiliaries
Steam Generator Pressure vs. Time

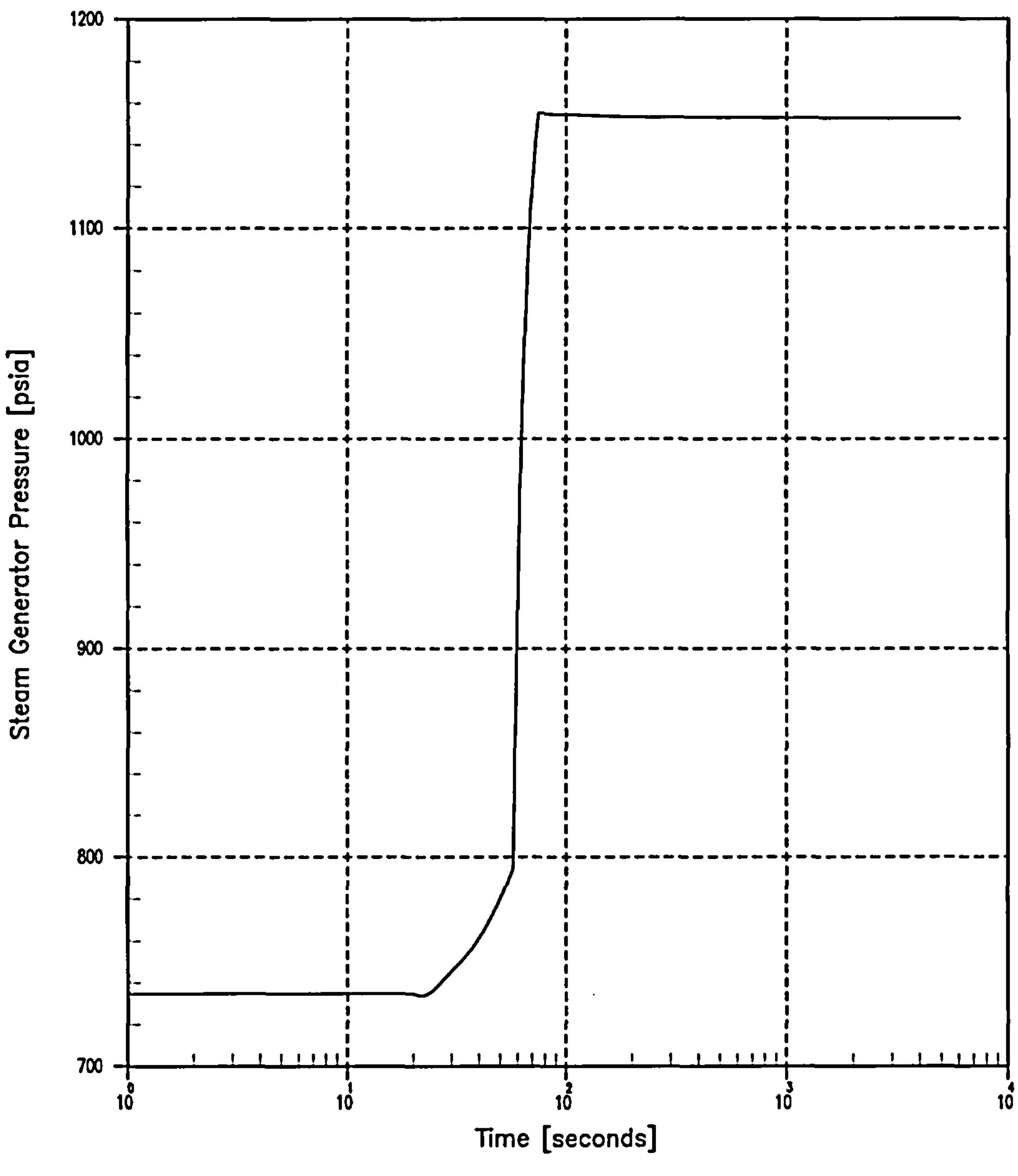


**Figure 14.1.12-5**

Loss of AC Power to the Plant Auxiliaries
Steam Generator Mass vs. Time

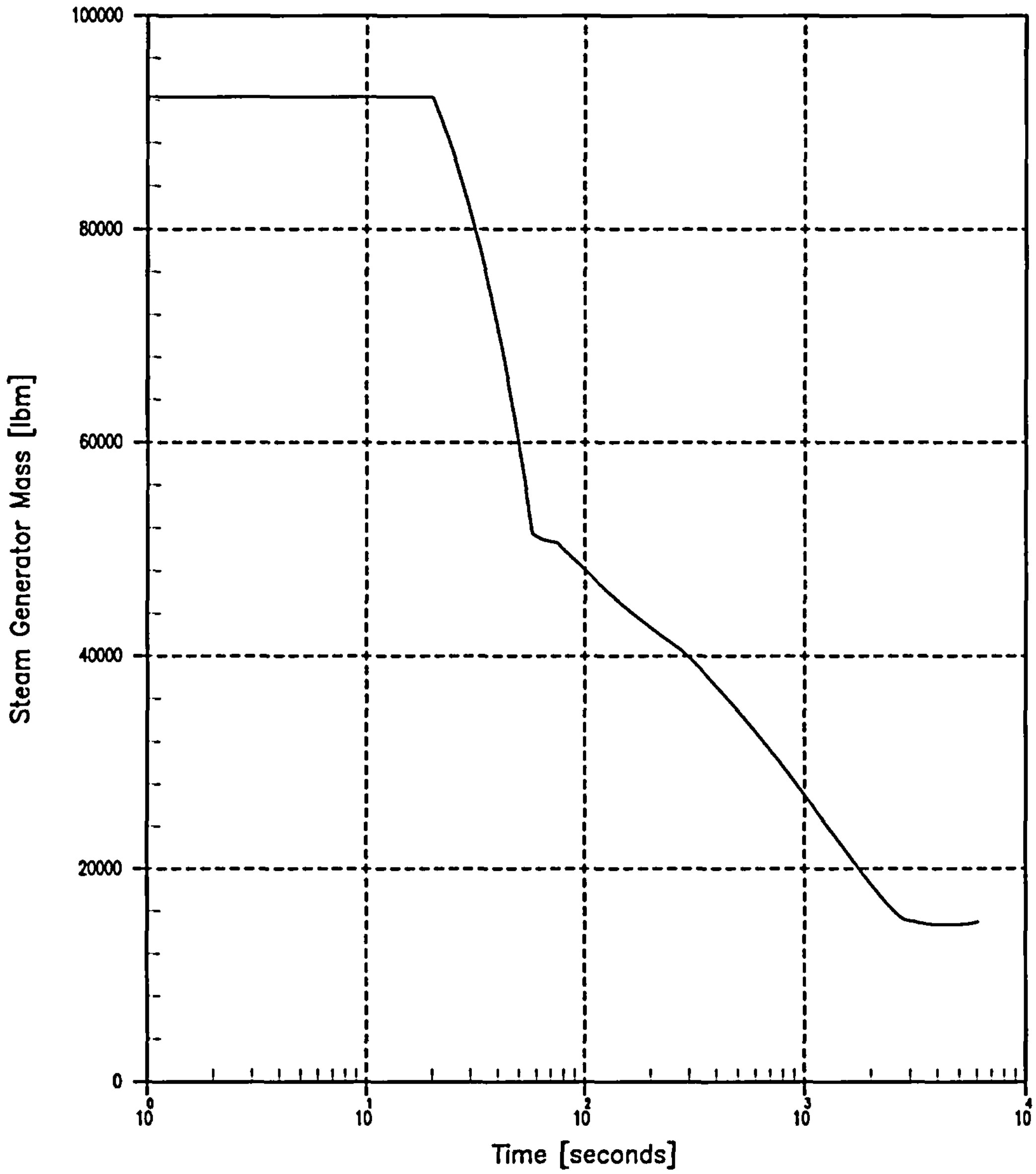


**Figure 14.1.12-6**

ATTACHMENT 6

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

**November 5, 2003**

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Responses to Requests for Additional Information and Supplemental Information Regarding LAR 195

Revised List of Regulatory Commitments

3 Pages to Follow

## LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by NMC in this document (#16), including those made in LAR 195. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

| | Commitment Description | Due Date/Event |
|---|---|---|
| 1 | Section 7.1.2 of attachment 4 states that an evaluation of the thermal and hydraulic safety analyses for the Framatome ANP fuel demonstrates that the DNBR design basis is met for the Framatome ANP fuel in Cycle 26. This evaluation has been performed for cycle 26 reload safety evaluation, the COLR, and the associated 10 CFR 50.59. All documents will be revised, as appropriate, for the stretch power uprate to address Framatome fuel DNBR design basis. | Prior to implementation of the stretch power uprate. |
| 2 | Sections 8.3.3.3 and 8.3.7.3 of attachment 4 state the increase in flow rate and velocities, as well as the changes in operating pressures and temperatures, will be incorporated into the KNPP FAC program as part of the power uprate implementation. The FAC program models will be updated prior to the next program inspections scheduled for the next refueling outage. | Prior to the next scheduled refueling outage. |
| 3 | Section 8.3.3.3, 8.3.15, 8.6.4, 8.6.5, and 8.6.6 of attachment 4 all refer to the feedwater control valve trim modification. This modification was completed during the R26 refueling outage. | Complete. |
| 4 | Section 8.3.3.3 of attachment 4 describes the following in regards to the feedwater heaters:<br><br>a. Initial inspection and analyses will establish a baseline prior to the stretch uprate implementation. These baseline inspections and analyses were completed during the R26 refueling outage.<br><br>b. An inspection and monitoring program will be established to monitor potential heater degradation at the stretch power uprate conditions. An inspection program will be developed based on the baseline inspection results and using programs and processes in place at KNPP. This will be completed prior to the next refueling outage. | <br><br>a. Complete.<br><br>b. Prior to the next refueling outage. |
| 5 | Sections 8.3.8.4 and 8.4.2.2.1 of attachment 4 state that a cycle-specific heat load calculation will be performed prior to each refueling outage to determine the adequacy of spent fuel pool cooling capability. Reactor engineering refueling procedures have already incorporated this confirmatory calculation as a requirement. The new requirement will administratively control the in core hold time of the fuel after shutdown to ensure the requirements are met. | Complete. |

| | Commitment Description | Due Date/Event |
|---|---|---|
| 6 | Instrument and Control System Setpoint changes were summarized in sections 8.3.9.3 and section 8.3.15 of attachment 4. These recommended changes included CST level setpoints, first stage turbine pressure, and turbine overspeed trip settings. The turbine overspeed trip setting changes were completed during the R26 refueling outage. The other recommendations will be reviewed by the plant staff and implemented as appropriate. | Prior to stretch uprate implementation. |
| 7 | Piping and pipe support evaluations concluded in section 8.4.4 that the systems remain acceptable assuming resolution of open items. Open items remained on the following systems: service water and component cooling water. These open items will be resolved. | Prior to stretch uprate implementation. |
| 8 | Modifications to the steam generator level control system were recommended to support the stretch uprate as described in Sections 8.6.5 and 8.6.6 of attachment 4. The level control changes were completed during the R26 refueling outage. | Complete. |
| 9 | Evaluation of the EQ equipment inside containment affected by the higher containment EQ long-term temperature profile will be performed per Sections 8.9.3 and 8.9.4 of attachment 4. | Prior to implementation of stretch power uprate. |
| 10 | The KNPP EQ Plan will be updated, as appropriate, to reflect power uprate evaluations in accordance with Section 8.8.4.3 of attachment 4. | Prior to implementation of stretch power uprate. |
| 11 | For those components where the HELB temperatures exceeded the equipment qualification temperature, the EQ equipment required for HELB outside containment will be qualified per Section 8.9.3 and 8.9.4 of attachment 4. | Prior to implementation of stretch power uprate. |
| 12 | Plant procedures will be revised as appropriate to accommodate the stretch power uprate. Procedure changes were committed to in Attachment 1, Section 2.6 and in Attachment 4, Section 8.7.2. Emergency, abnormal, and operating procedures that are entered due to a LONF event or have AFW TS requirements will be changed as appropriate to reflect the new TS requirements for the AFW system. | Prior to implementation of stretch power uprate. |
| 13 | New TS requirements, revised procedures, and any control room changes due to the stretch power uprate will be reviewed by training for determination of being included in the operator training program. This was committed to in Attachment 1, Section 2.6. | Prior to implementation of stretch power uprate. |
| 14 | Attachment 1, Section 2.6 stated that setpoint changes for reactor protection and control inputs, alarms, computer constants, and embedded values, will be updated consistent with operation at 1772 MWt. Power range nuclear instruments will be recalibrated and checked based on a secondary heat balance. | Prior to implementation of stretch power uprate |
| 15 | Sections 9.2.4.1.7 and 9.2.4.2.5 of attachment 4 describe recommended modifications to the high pressure turbine cylinder joint bolting and the low-pressure turbine coupling bolts. These modifications were completed during the recent R26 refueling outage. | Complete. |

| | Commitment Description | Due Date/Event |
|---|---|---|
| 16 | The response to RAI question #57 to the stretch power uprate submittal stated degraded voltage and thrust calculations for MOV operators outside containment were reviewed for impact of the uprated post accident temperatures, and will be revised, as required, prior to implementation of the 6% Stretch Power Uprate. | Prior to implementation of stretch power uprate. |