NMC
Committed to Nuclear Excellence

Kewaunee Nuclear Power Plant
Operated by Nuclear Management Company, LLC

NRC-03-057 10 CFR 50.90

May 22, 2003

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

KEWAUNEE NUCLEAR POWER PLANT
DOCKET 50-305
LICENSE No. DPR-43
LICENSE AMENDMENT REQUEST 195, APPLICATION FOR STRETCH POWER UPRATE FOR KEWAUNEE NUCLEAR POWER PLANT

References:

1) Letter from Mark E. Warner to Document Control Desk, "Revision to the Design Basis Radiological Analysis Accident Source Term," dated March 19, 2002 (TAC No. MB4596).

2) Letter from Mark E. Warner to Document Control Desk, "License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications Changes for Use of Westinghouse VANTAGE+ Fuel," dated July 26, 2002 (TAC No. MB5718).

3) Letter from NRC to Mark Warner, "Kewaunee Nuclear Plant - Review of Submittals Supporting Future Measurement Uncertainty Recapture Power Uprate (TAC Nos. MB4596, MB5717, and MB5718)," dated August 16, 2002.

4) Letter from NRC to NMC, LLC, "Meeting Summary of August 8, 2002 Between the NRC Staff and the Nuclear Management Company, LLC Concerning Upcoming Power Uprate and Associated Submittals for the Kewaunee Nuclear Power Plant," dated August 20, 2002.

5) Letter from Thomas Coutu to Document Control Desk, "Kewaunee Nuclear Power Plant Request for Use of GOTHIC 7 in Containment Design Basis Accident Analysis," dated September 30, 2002 (TAC No. MB 6408).

6) Letter from Thomas Coutu to Document Control Desk, "License Amendment Request 193, Measurement Uncertainty Recapture Power Uprate for Kewaunee Nuclear Power Plant," dated January 13, 2003 (TAC No. MB 7225).

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7) Letter from John G. Lamb to Mr. Thomas Coutu, "Kewaunee Nuclear Power Plant - Issuance of Amendment (TAC No. MB 5718)," dated April 4, 2003 (regarding use of Westinghouse 422 VANTAGE + nuclear fuel with PERFORMANCE + features).

8) Letter from John G. Lamb to Mr. Thomas Coutu, "Kewaunee Nuclear Power Plant - Issuance of Amendment Regarding Implementation of Alternate Source Term (TAC No. MB 4596)," dated March 17, 2003.

In accordance with the requirements of 10 CFR 50.90, Nuclear Management Company, LLC (NMC) requests amendments to the operating license and the plant Technical Specifications (TS) for the Kewaunee Nuclear Power Plant (KNPP). The license amendment request (LAR) would increase the licensed reactor core power level by six percent from 1673 megawatts thermal (MWt) to 1772 MWt. The requested increase in licensed rated power (RP) is the result of a stretch power uprate. It is important to note that this request for an increase in power starts at an uprated power level of 1673 MWt. The power level of 1673 MWt is based on a measurement uncertainty recapture (MUR) power uprate LAR (reference 6). Reference 6 is currently under review by the Nuclear Regulatory Commission (NRC) with an expected approval date of June 2003. Combined with the MUR power uprate, the KNPP total licensed rated power will be uprated from its original power of 1650 MWt to 1772 MWt, a total increase of 7.4 percent. However, the enclosed LAR relates only to the six percent stretch uprate. The six percent uprate meets the NRC's definition of a stretch uprate since it is less than seven percent and involves only minor modifications and changes to instrumentation settings (no major plant modifications).

It is important to note that this power uprate LAR relies in part upon four previous submittals by the NMC. The NRC has approved the first two submittals in the list below. However, the third and fourth submittals require NRC approval for the stretch power uprate. These submittals are:

1) the alternate source term (AST) methodology for design basis radiological accident analysis (references 1 and 8),
2) the fuel transition (references 2 and 7),
3) the 1.4 percent MUR power uprate (reference 6), and
4) the GOTHIC 7 computer code (reference 5).

The radiological accident analyses described in the approved AST submittal (references 1 and 8) were performed at a core power of 1650 MWt with a two percent uncertainty. The radiological accident analyses were reperformed using the approved AST methodology and assuming a reactor core power level of 1772 MWt with 0.6 percent uncertainty. Technical evaluation summaries of the reanalysis of the radiological accident analyses have been included in attachment 4, Section 6.7, for completeness.

The majority of the KNPP Updated Safety Analysis Report (USAR) Chapter 14 accident analyses were described in the fuel transition LAR (reference 2) and the NRC Safety Evaluation Report (SER) (reference 7). These analyses support the stretch uprate since they were performed at core powers at or in excess of 1772 MWt. The NRC reviewed these accident analyses for the fuel transition as described in reference 7. The exceptions to this are the loss of normal feedwater (LONF) event, the anticipated transients without scram (ATWS), the containment integrity analyses, the steam generator tube rupture thermal/hydraulic analysis, and the main steam line break consequences. These analyses, with the exception of LONF are included in Section 6 of attachment 4 to this letter.

A new LONF analysis performed at 1772 MWt has been included in attachment 4, Appendix 6A. Additionally, all accident analyses performed at 1772 MWt for support of the fuel transition are included in summary form in attachment 4, Section 6.0. The KNPP containment integrity analyses, described in attachment 4, Section 6.4, were performed using the GOTHIC 7 computer code. The application of this code for KNPP is described in the reference 5 submittal to the NRC. The NRC approval of the GOTHIC 7 code for use at KNPP is expected in August 2003.

Therefore, the starting point of this amendment is the assumption that the AST methodology, the fuel transition, and the MUR power uprate have been implemented and that the GOTHIC 7 containment analysis method has been approved. Based on this statement, NMC hereby proposes only those license and TS changes that are required for the six percent increase in rated power.

Several attachments to this letter support the KNPP stretch power uprate. The attachments are summarized in the table below:

Attachment	Content Description
1	A description and assessment of the stretch power uprate including: description, background, proposed license and TS changes, technical assessment, a no significant hazards consideration, and environmental considerations.
2	Facility Operating License, TS, and TS bases pages marked up to show the proposed changes.
3	Revised (clean copies) Facility Operating License, TS, and TS bases pages.
4	One copy of WCAP-16040-P, "Power Uprate Project, Kewaunee Nuclear Power Plant, NSSS and BOP Licensing Report," dated February 2003 (Proprietary). This report contains summaries of the technical evaluations performed for the combined MUR and stretch power uprates.
5	One copy of WCAP-16040-NP, "Power Uprate Project, Kewaunee Nuclear Power Plant, NSSS and BOP Licensing Report," dated February 2003 (Non-proprietary). This attachment contains the non-proprietary version of attachment 4.
6	Westinghouse authorization letter, CAW-03-1603, an accompanying affidavit, proprietary information notice, and copyright notice for attachment 4.
7	List of regulatory commitments associated with this LAR.

As attachment 4 contains information proprietary to Westinghouse Electric Company, it is supported by an affidavit signed by Westinghouse, 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.

NMC requests approval of this proposed amendment by February 23, 2004 with an implementation period of 90 days. This would support a mid-cycle power uprate for KNPP prior to the summer months of 2004. This date allows KNPP to take advantage of the economic benefits of the power uprate as soon as possible. It should be noted that the plant does not require this amendment to allow continued safe, full power operation.

The NMC has determined that the information for the proposed amendments does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendments meet the categorical exclusion requirements of 10 CFR 51.22(c)(9) and an environmental impact appraisal need not be prepared.

In accordance with 10 CFR 50.91, a copy of this application, with only the non-proprietary attachments, is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on May 22, 2003.

for

Thomas Coutu
Site Vice-President, Kewaunee Plant

LMG

Attachments:

1. Description of the Proposed Changes, Safety Evaluation, Significant Hazards Determination, and Statement of Environmental Considerations
2. Strike-Out Pages for License, Technical Specifications, and Bases
3. Revised Pages for License, Technical Specifications, and Bases
4. WCAP-16040-P, "Power Uprate Project, Kewaunee Nuclear Power Plant, NSSS and BOP Licensing Report," dated February 2003 (Proprietary)
5. WCAP-16040-NP, "Power Uprate Project, Kewaunee Nuclear Power Plant, NSSS and BOP Licensing Report," dated February 2003 (Non-Proprietary)
6. Westinghouse Authorization Letter, CAW-03-1603, an Accompanying Affidavit, Proprietary Information Notice, and Copyright Notice for Attachment 4
7. List of Regulatory Commitments

cc: US NRC, Region III without attachments 4 and 5
US NRC Senior Resident Inspector without attachments 4 and 5
Electric Division, PSCW without attachment 4

ATTACHMENT 1

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

May 22, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 195

Description of the Proposed Changes, Safety Evaluation, Significant Hazards Determination, and Statement of Environmental Considerations

1.0 Introduction and Background

Nuclear Management Company, LLC (NMC), proposes to amend the Facility Operating License and technical specifications (TS) to increase the licensed rated power (RP) level for the Kewaunee Nuclear Power Plant (KNPP). The KNPP is currently licensed to operate at a maximum RP of 1650 megawatts thermal (MWt). Approval was requested in reference 3 to increase the licensed RP by 1.4 percent to 1673 MWt (measurement uncertainty recapture (MUR) power uprate request). The reference 3 power increase is currently under review with an expected Nuclear Regulatory Commission (NRC) approval date in June 2003. Approval is now being requested for an increase in the licensed RP by an additional six percent from 1673 MWt to 1772 MWt. When combined with the application for the MUR power uprate, the total core power increase is 7.4 percent. However, this license amendment request relates only to the six percent stretch uprate. The six percent uprate meets the definition of a stretch uprate since it is less than seven percent and involves only minor modifications and changes to instrumentation settings (no major plant modifications).

KNPP was originally designed with equipment and systems capable of accommodating operating conditions above the original licensed power rating. The original NRC Safety Evaluation Report (SER) for the KNPP dated July 24, 1972 stated, "The reactor is expected to be capable of an ultimate output of 1721.4 MWt, however, and our evaluation of the containment, the engineered safety features and the accident analyses has been performed for this maximum power of 1721.4 MWt." Additionally, the SER stated, "Our evaluation of the thermal, hydraulic and nuclear characteristics of the reactor core was for a power rating of 1650 MWt. Before operation at any power level above 1650 MWt is authorized, the regulatory staff will perform a safety evaluation to assure that the core can be operated safely at the higher power level." Continuing improvements in the analytical techniques, measurement instrument accuracies, plant thermal performance, and fuel and core designs have resulted in an increased margin between the safety analyses results and the licensing limits. These available margins, combined with the excess margin in the as-designed equipment, system and component capabilities, provide KNPP with the potential for an increase in thermal power rating of 7.4 percent (a 1.4 percent MUR and a six percent stretch power uprate) over the original licensed rated power of 1650 MWt without major nuclear steam supply system (NSSS) or balance of plant (BOP) hardware modifications and with no significant increase in the hazards presented by the plant.

The NMC has evaluated the impact of a six percent stretch power uprate from 1673 MWt to 1772 MWt for the applicable systems, structures, components, and safety analyses at KNPP. The results of this evaluation are described in attachment 4 of this letter, WCAP-16040-P, "Power Uprate Project, Kewaunee Nuclear Power Plant, NSSS and BOP Licensing Report." The KNPP stretch power uprate analyses and evaluations were performed consistent with the guidelines set forth in the Westinghouse's WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," January 1983. The WCAP was submitted to the NRC for review in a letter from Westinghouse to the NRC dated February 11, 1983. This methodology, although not formally reviewed and approved by the NRC, has been used successfully in several NRC-approved power uprate submittals.

The stretch uprate program included the reanalysis or evaluation of the loss of normal feedwater accident, the anticipated transients without scram (ATWS), containment integrity analyses, the steam generator tube rupture thermal/hydraulic analysis, the main steam line break consequences, and radiological accident analyses. The remaining Chapter 14 analyses were reviewed and approved with the fuel transition (references 2, 6, and 7) and the summaries have been included in attachment 4, Section 6.0 for completeness. The stretch uprate program also included evaluation of major NSSS components (e.g., reactor pressure vessel, pressurizer, reactor coolant pumps, and steam generators), BOP components (e.g., turbine, generator, and condensate and feedwater pumps), and major systems and sub systems (e.g., safety injection, auxiliary feedwater, residual heat removal, electrical distribution, emergency diesel generators, containment cooling). The NSSS component analyses performed at stretch uprate conditions of 1772 MWt were included in the 1.4 percent MUR uprate (references 3 and 10). Control systems (e.g., rod control, pressurizer pressure and level, turbine overspeed, steam generator level, and steam dump) have been evaluated for operation at uprated power conditions. Reactor protection trip and engineered safety features (ESF) actuation setpoints have been assessed for the uprated power conditions. The results of all of the above analyses and evaluations have yielded acceptable results and demonstrate that all design basis acceptance criteria will continue to be met during uprated power operations. These evaluations are contained in attachment 4 to this letter.

To accommodate operation at the uprated power level, two minor plant modifications were necessary. These modifications were changes to the valve trim in the feedwater control valves to accommodate higher feedwater flow rates and bolting changes on the turbine generator to handle higher operating stresses associated with uprate operating conditions. These modifications are discussed in attachment 4, Sections 8.3.3.3 and 9.2.4, respectively. Both of these modifications were completed during the spring 2003 refueling outage. Additionally, there will be control and protection setpoint changes to accommodate the stretch power uprate.

2.0 Technical Assessment of the Change in Rated Power

Attachment 4 describes the analyses performed to support the six percent stretch power uprate. These include accident analyses (most of which were previously submitted during the fuel transition, references 2 and 7), system and component analyses (some of which were submitted during the 1.4 percent MUR power uprate, references 3 and 10), and radiological analyses. Additional items outside of those evaluated in attachment 4 will be discussed below in detail. These include:

1. An assessment of radiological analysis for the technical support center (TSC) at the uprated conditions.

2. A description of the changes to the loss of normal feedwater (LONF) accident analysis assumptions for equipment operability and basis of these changes to the KNPP licensing basis.

3. An evaluation of containment sump blockage that KNPP was requested to answer as part of the response to a fuel transition request for additional information (RAI) (reference 6).

4. An assessment of the containment cooling TS changes that are based on the uprated containment integrity analysis.

5. Closure of the open item relative to re-evaluation of fuel clad stress values documented in Attachment 4, Section 7.2.3.

6. An assessment of the human factors and instrument and control (I&C) changes expected for the stretch power uprate.

2.1 Technical Support Center

Section 8.8.3.3 of attachment 4 states that, "KNPP is currently going forward with the implementation of AST methodology and the dose estimates in the technical support center and control room, based on power uprate conditions and AST methodology." Amendment No. 166 and its associated SER (reference 4) regarding the use of the AST methodology at KNPP were issued on March 17, 2003. This amendment revised the radiological consequence analyses for the KNPP design basis accidents to implement the AST methodology. Part of the submittal was the analysis of the control room dose. The NRC staff reviewed the analyses, performed confirmatory assessments of the radiological consequences of the postulated design basis accidents (DBAs), and documented acceptance in the SER. The TSC dose was not analyzed as part of the submittal.

The AST methodology as requested and approved (references 1 and 4) fell under Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," paragraph 1.2.2, *Selective Implementation*. Therefore, new applications of the selected AST characteristics would not require prior NRC approval unless stipulated by 10 CFR 50.59. Reactor core fission product inventory available for release to containment as postulated in the request for Amendment No. 166, assumes maximum full power operation at a core power of 1650 MWt with two percent calorimetric uncertainty. The DBAs that were part of Amendment 166 were re-performed for the six percent stretch power uprate at a core power of 1772 MWt (with 0.6 percent uncertainty) using the approved AST methodology. The radiological consequences evaluation summaries for the uprate are located in Section 6.7 of attachment 4.

NUREG-0737, "Clarification of TMI Action Plan Requirements," Section III.A.1.2, requires that the TSC shall be habitable to the same degree as the control room for postulated accident conditions. The original evaluation results for the TSC dose and control room dose calculations indicated that the control room dose was more limiting than the TSC dose. Since the control room radiological analyses at 1772 MWt using the AST methodology concluded the control room doses were acceptable, it could be concluded the TSC doses would also remain acceptable for power uprate condition based on the earlier comparison of the original dose calculation results. However, the control room model used in the AST radiological analyses was not consistent with the model used for the original control room evaluation. Therefore, an evaluation was necessary compare the control room models and to demonstrate that the control room dose remained more limiting than the TSC dose at uprated powers.

The AST evaluation of the control room models a higher than original unfiltered in-leakage (200 cfm versus 55 cfm) and a higher recirculation flow (2250 cfm versus 1121 cfm). Using these two changed parameters, along with the unchanged parameters (filter efficiency, volume, TSC design information, etc.), a comparison was made between the control room models and the TSC. The comparison evaluation confirmed that the TSC doses for the 7.4 percent power uprate would be less than those calculated for the control room. Therefore, the TSC continues to meet habitability and dose requirements (i.e., less limiting than control room dose) at uprated conditions.

2.2 Loss of Normal Feedwater Evaluation

A LONF results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the reactor is not tripped during this accident, core damage could possibly occur from a sudden loss of heat sink. If an alternative supply of feedwater is not supplied to the steam generators, residual heat following the reactor trip heats the coolant to the point where water relief from the pressurizer occurs. Subsequent loss of water from the reactor coolant system could conceivably lead to core damage.

This transient was reanalyzed for 1772 MWt core conditions. A LONF analysis at 1772 MWt was originally transmitted to the NRC in reference 2 but then later retracted by attachment D of reference 6. The reference 3 submittal referred to the current LONF analysis performed at 1650 MWt with an additional two percent uncertainty. The new LONF analysis at 1772 MWt is submitted with this letter and is located in attachment 4, Appendix 6A. Based on the results of the LONF at the uprated core power of 1772 MWt, new TS requirements for AFW train operability have been created. Auxiliary feedwater trains are defined in current TS basis for 3.4.b on page TS B3.4-1. The AFW train operability requirements are discussed in the next paragraphs and the proposed changes to TSs are described in Section 3.6 of this attachment.

For operation at or below 1673 MWt, the current LONF analysis performed at 1650 MWt with two percent calorimetric uncertainty (equivalent to 1673 MWt with 0.6 percent uncertainty in reference 3) remains acceptable. This current LONF analysis at a total core power of 1683 MWt, uncertainty included, demonstrates that only one AFW pump train at minimum flow capability is necessary to prevent water relief from the pressurizer. Therefore, KNPP TS will clarify that at or below 1673 MWt, two AFW trains must be operable. Two trains must be operable to meet accident analysis assumptions for AFW flow and the limiting single failure criteria of one AFW pump.

For operation above 1673 MWt, the LONF analysis performed at 1772 MWt with 0.6 percent uncertainty must be met. The LONF analysis at a total power 1782.6 MWt, including uncertainty, requires two AFW pumps for heat removal capability and prevention of pressurizer overfill. Requiring three AFW trains to be operable at all times above a power level of 1673 MWt provides this capability and allows for the limiting single failure of one AFW pump. However, this is a change to the current KNPP licensing basis and requires changes to AFW system TSs.

The proposed TS changes to support the uprated LONF analysis are described in Section 3.6 of this attachment. The LONF analysis results are summarized in attachment 4, Appendix 6A.

2.3 *Containment Sump Blockage Prevention*

The NRC RAI 55 to LAR 187 (reference 6) required information related to the containment emergency sump. This RAI asked for a description of the containment emergency sump configuration, including the screen design, showing that an adequate water source is provided to the emergency core cooling systems (ECCS) during operation after initiation of recirculation from the emergency sump. The NMC response to this RAI stated, "This RAI is related to power uprate only and will be addressed in the Stretch Power Uprate LAR." This specific requested information is not addressed in attachment 4 and is addressed below.

Nothing within the containment emergency sump configuration or ECCS is changed as a result of the power uprate. The power uprate does not introduce any new sources of debris to the containment. The KNPP NSSS auxiliary tanks, heat exchangers, pumps, and valves are acceptable for the uprating conditions, and there is no change to the auxiliary systems operating conditions identified as a consequence of the uprating. The system and component evaluation results for the uprating are consistent with, and continue to comply with, the current KNPP licensing basis and acceptance requirements.

Containment Sump B is situated below the 592' elevation inside the containment and is covered by two conical strainers. Flow out of the sump to residual heat removal (RHR) pumps for ECCS recirculation must pass through these conical strainers. The screens of the strainers are sized to prevent any particles with a mean diameter greater than 1/8 inch from entering the RHR pumps. Previous evaluations performed at KNPP for the effect of failed unqualified containment coatings on sump performance do not change based on the uprate. Concerning insulation and construction materials, the NRC staff has investigated the buoyancy, transport, and head loss characteristics of reflective metallic insulation and construction materials and the results are summarized in NUREG/CR-3616, "Transport and Screen Blockage Characteristics of Reflective Metallic Insulation Materials." The results showed that thin metallic foils could be transported at low flow velocities and that flow blockage could occur at the lower portion of the screen. Since the KNPP screens are elevated and are approximately four feet tall, insulation and construction materials would cause little, if any, blockage of the screens.

Original manufacturer design information states each screen is designed for 1.5 feet of head loss. The manufacturer design head loss of 1.5 feet incorporates a significant margin for screen blockage. The manufacturer's calculation for Kewaunee's sump screens supports the head loss assumption made in the original sump screen documentation. The manufacturer's calculation assumes significant screen blockage with the screens fully submerged. The calculations indicate that the blockage must be over 85 percent before the head loss would become 1.5 feet. Also, there is no increase in sump flow rates or requirements which would affect strainer delta pressure.

In summary, the power uprate does not change the containment sump design, flow rates, or the current calculations for the sump screens. Current calculations continue to validate the acceptability of the design. Therefore, for the power uprate, there is no change in the effect that potential debris accumulation would have on the effective water pressure and flow rate available to the ECCS pump suctions after initiation of recirculation. The containment sump will continue to function as designed and all ECCS flows remain acceptable at the power uprate conditions.

2.4 Containment Cooling System Change

This TS change, described in Section 3.4 of this attachment, removes the Limiting Condition for Operation (LCO) that allowed both containment fancoil unit (CFCU) trains to be out of service for 72 hours provided both containment spray trains remain operable. The basis of this change is the new containment integrity analysis for the uprated condition of 1772 MWt. The NMC determined in this analysis that KNPP would exceed heat-removal capability of the containment spray (CS) pumps during post-accident conditions at the uprated power. An effect of this is that the existing LCO of the containment fancoil units that relies on full-capacity heat removal capability of the containment spray pumps must be deleted.

Design considerations combine containment cooling equipment into two trains, each train consisting of two containment fan cooling units (CFCU) and one containment spray pump. Each train is fully capable of controlling post-accident containment temperature and pressure, and is electrically and mechanically separated from the opposite train. Each train will remain capable of its design safety function after power uprate. There is no normal plant operating condition in which the CFCU and containment sprays would be operated in the configuration implied by the deleted LCO.

Deletion of the LCO is conservative, consistent with existing design basis, and conforms to safety analysis acceptance criteria. Therefore, the NMC concludes that deletion of the LCO is appropriate and necessary in consideration of power uprate requirements, that it does not alter existing margins to safety, and that it will not create a circumstance adverse to safe operation.

2.5 WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel"

Attachment 4, Section 7.3.2, "Conclusions," states that, "Pending approval of WCAP-10125 by the NRC, subsequent re-evaluation of the stress values will be performed to confirm the proposed clad stress criterion is met." This same commitment to re-evaluate the stress values was made in the NMC submittal for the fuel transition (reference 2). The commitment refers to Addendum 1 to WCAP-10125-P-A, which had previously been submitted to the NRC as Addendum 2 to WCAP-12488. After submittal of WCAP-12488 by Westinghouse, the NRC requested that it be resubmitted as Addendum 1 to WCAP-10125 (reference 9).

NRC approval of Addendum 1 to WCAP-10125-P-A was obtained on April 14, 2003 (reference 11). Evaluations of the fuel cladding have been performed in accordance with WCAP-10125 and have produced acceptable results. In a letter to the NRC dated March 21, 2003 (reference 9), the NMC confirmed that these evaluations had been completed acceptably. Reference 9 closed the commitment for the fuel transition and closes this open item in Section 7.3.2 of attachment 4.

2.6 Human Factors and I&C Changes

This section addresses human factors and I&C changes as suggested on the NRC Power Uprate website guidance.

<u>Changes in Emergency and Abnormal Operating Procedures</u>

Changes to emergency and abnormal operating procedures are expected to be minor. The changes would include setpoint changes. Emergency and abnormal operating procedures that are entered due to a LONF event will be changed to reflect the new TS requirements for the AFW system.

<u>Changes to Operator Actions Sensitive to Power Uprate</u>

There are no new operator actions, additional response times needed, or reduced times available in emergency or abnormal operating procedures as a result of the stretch power uprate. Any emergency or abnormal operating procedures regarding AFW TS requirements will be changed prior to implementation of the stretch power uprate. New TS requirements and the basis for them will be reviewed in the operator-training program.

Changes to Control Room Controls, Displays, and Alarms

The plant process computer screen (PPCS), which displays the UFMD/UTM correction factors, UFMD OPERATING LIMIT, and RTO OPERATING LIMIT (discussed in detail in the MUR uprate LAR, references 3 and 10), will be updated for operation at 1772 MWt. In addition, the CROSSFLOW computer program constants will be updated consistent with operation at 1772 MWt. These updates will also change the alarm setpoints associated with the UFMD (ultrasonic flow measuring device) operational limits. UFMD abnormal and alarm procedures will be updated as necessary.

In addition to the PPCS screen changes mentioned above, PPCS computer constants and embedded values will be updated consistent with the new licensed rated power of 1772 MWt. Computer constants, such as full power delta T and licensed rated power level, and programs, such as the RTO program, will be updated.

Changes on the Safety Parameter Display System

No changes to the safety parameter display system are expected as a result of the stretch power uprate.

Changes to the Operator Training Program and the Control Room Simulator

The operator training program will review the changes made to the technical specifications and procedures made as a result of the stretch power uprate. The Operations Training Department will determine the extent of training based on the changes prior to the stretch power uprate implementation. The plant simulator will be modified to reflect any control room changes due to power uprate.

Suitability of Existing Instruments

No modifications to existing instrumentation and controls are required for the stretch power uprate other than certain setpoint, PPCS constants, and PPCS program changes.

RPS and ESFAS Instrumentation Trip Setpoint and Allowable Values

The instrument changes required to support the stretch power uprate (increase licensed power from 1673 MWt to 1772 MWt) are:

- Changing the full power ΔT_0 inputs to the overtemperature delta T (OTDT) and the overpower delta T (OPDT) setpoints to the predicted value based on best estimate evaluations for the stretch uprated power (1772 MWt) condition.

- Recalibration of the power range nuclear instruments (NIs) will be performed based on scaling the 1673 MWt full power NI currents to the 1772 MWt value. Following recalibration of the power range NI currents to the new projected full power values, the NI calibration will be checked based on a secondary heat balance calculated for the new 100 percent licensed power level. Once the power range NIs have been adjusted to the appropriate percent power for the new licensed power level, all the power range reactor trips, rod stops, and permissives (P7, P8, and P10) based on percent power will function at the appropriate value.

- Rescaling the turbine first stage pressure inputs to the P13 permissive that provides the turbine first stage pressure input to permissive P7.

TS Changes Related to the Power Uprate

Changes to the technical specifications related to the stretch power uprate are described in Section 3.0 of this attachment.

3.0 Description of License and Technical Specification Changes including No Significant Hazards Determination

The proposed license amendments will revise the Kewaunee Nuclear Power Plant (KNPP) Facility Operating License and the technical specifications (TS) to increase the licensed rated power (RP) by six percent from 1673 megawatts thermal (MWt) to 1772 MWt. The proposed changes are described in detail below and are also indicated on the marked up and clean copy Operating License and TS pages in attachments 2 and 3. The Operating License and TS changes have been grouped and each group will be evaluated with respect to the criteria of 10 CFR 50.92. Corresponding TS bases changes are being made to reflect changes to the TS.

3.1 Power Level Changes

Description of Proposed Power Level Changes

Paragraph 2.C. (1), "Maximum Power Level," of the operating license, DPR-43, has been revised to authorize operation at reactor core power levels not in excess of 1772 MWt.

Paragraph TS 1.0.m, RATED POWER, has been revised to reflect the increase from 1673 MWt to 1772 MWt.

Significant Hazards Determination for Proposed Power Level Changes

The KNPP stretch power uprate evaluations in attachment 4 demonstrated that this increased core power still allows safe operation of the plant and will not affect the health and safety of the public. Based on the evaluations at the proposed uprate conditions the following conclusions can be reached with respect to 10 CFR 50.92. The proposed change will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The stretch uprate evaluations performed included performance of accident analyses at uprated power parameters using approved methodologies. Results of these analyses continue to meet the event acceptance criteria. An evaluation of components and systems, including interface and control systems, that could be affected by the change in power level, were performed for the stretch power uprate. Components and systems will continue to function as designed and performance requirements for these systems will continue to be met. Additionally, the proposed change in power level was not found to initiate any accident, and therefore, does not increase the probability of an accident.

Dose consequences were evaluated using the uprated power parameters. Acceptance criteria continue to be met. Therefore, the change also does not increase the consequences of an accident.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The change has no adverse effect on any safety related system and does not change the performance or integrity of any safety related system. Additionally, no new safety related equipment is being added or changed as a result of this proposed change in power. Therefore, the possibility of a new or different kind of accident is not created.

3. Involve a significant reduction in the margin of safety.

   All analyses supporting the proposed uprated power condition continue to meet the appropriate acceptance criteria. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Therefore, there are no significant hazards associated with the changes in rated power level.

3.2 Proposed Safety Limit Change

Description of Proposed Safety Limit Change

Paragraph TS 2.1.c, regarding peak fuel centerline temperature, has been revised to increase the peak fuel centerline temperature from < 4700°F to < 5080°F. Additionally, the following text has been inserted at the end of TS 2.1.c, "decreasing by 58°F per 10,000 MWD/MTU of burnup."

Significant Hazards Determination for Proposed Safety Limit Change

The basis for this change is the recent change to Westinghouse fuel. The 5080°F minus 58°F per 10,000 MWD/MTU limit has been the fuel melt limit for Westinghouse fuel since the mid-1960s. This limit is an industry-accepted limit for Westinghouse fuel, backed by experimental data and Westinghouse technical evaluations. Westinghouse initially established an internal design criterion of 4700°F when assessing the fuel melt limit. However, Westinghouse found that in reality, the design criterion was not a flat value, but a decreasing value as documented in their evaluations. The experimental data is applicable to all uranium dioxide fuel, so it is also applicable to the Framatome fuel remaining in the core. Additionally, during KNPP's request for a core operating limits report (COLR) (reference 12), NMC discussed why the 4700°F remained an acceptable value as opposed to using a peak fuel centerline temperature adjusted for burnup (reference 13). The NMC is now changing this TS to be consistent with NUREG-1431 (reference 8). Based on these evaluations, the following conclusions can be reached with respect to 10 CFR 50.92. The proposed change will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change is an industry accepted safety limit applicable to the KNPP transition to Westinghouse fuel. Therefore, the change does not increase the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed change in fuel centerline temperature. The change has no adverse effect on the fuel or the performance or integrity of the fuel. Therefore, the possibility of a new or different kind of accident is not created.

3. Involve a significant reduction in the margin of safety.

   The proposed safety limit change is backed by technical evaluations performed by Westinghouse and experimental data. The limit is shown to be met as part of reload safety evaluations performed on a cycle specific basis. All applicable analyses supporting the proposed uprated power condition continue to meet the appropriate acceptance criteria. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Therefore, there are no significant hazards associated with the change in the safety limit.

3.3 Engineered Safety Feature (ESF) Setting Change

Description of Proposed ESF Setting Change

Table TS 3.5-1, "Engineered Safety Features Initiation Instrument Setting Limits," setting limit for 6, "High-High Steam Flow in a Steam Line Coincident with Safety Injection," has been changed from 4.5 X 10^6 lb/hr at 735 psig to 4.4 X 10^6 lb/hr at 735 psig.

Significant Hazards Determination for Proposed ESF Setting Change

The basis of the change to the high-high steam flow TS setting is the change in the analytical limit used in the high-high steam flow setpoint calculation. This calculation documented the total loop accuracy for the high-high steam flow actuation logic circuit and was used to determine the TS limit and the actual plant setting for the high-high steam flow trip. The new analytical limit in the setpoint uncertainty calculation corresponds with the maximum calibrated span of the plant's steam flow transmitters. The actual plant setting for high-high steam flow trip is within the loop calibrated span which ensures the trip will occur prior to reaching the analytical limit. Additionally, the accident analyses assume a much higher high-high steam flow trip (7.76 X 10^6 lb/hr), which causes the trip to actuate later in the accident analysis and is considered more conservative. Based on this, the following conclusions can be reached with respect to 10 CFR 50.92. The proposed change will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The stretch power uprate evaluations performed included performance of accident analyses. Results of the accident analyses have verified that the acceptance criteria continue to be met. Neither the change in the analytical limit nor the change in the TS setting limit changes how the system functions. Systems will continue to function as designed and system performance criteria will continue to be met. Dose consequences have also been evaluated at uprate conditions and doses remain within the appropriate acceptance criteria. Therefore, the change does not increase the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The change has no adverse effect on any safety related system and does not change the performance or integrity of any safety related system. Additionally, no new safety related equipment is being added or changed as a result of the proposed change in the high-high steam flow TS setting limit. Therefore, the possibility of a new or different kind of accident is not created.

3. Involve a significant reduction in the margin of safety.

   The results of the accident analyses demonstrate the acceptance criteria continue to be met. Systems will continue to function as designed and system performance criteria continue to be met. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Therefore, there are no significant hazards associated with the change in the high-high steam flow TS setting limit.

3.4 Proposed Containment Cooling Systems Change

Description of Proposed Containment Cooling Systems Change

Paragraph TS 3.3.c.1.A.3 (iii), which allowed both containment fancoil unit (CFCU) trains to be out of service for 72 hours provided both containment spray trains remain operable, has been deleted in its entirety and the subsequent item, (iv), renumbered as (iii).

Significant Hazards Determination for Proposed Containment Cooling Systems Change

Based on the power uprate containment integrity evaluations and the fact that deletion of the Limiting Condition for Operation (LCO) is conservative, the following conclusions can be reached with respect to 10 CFR 50.92. The proposed change will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   Removal of the LCO is conservative in that it eliminates relaxation of a design requirement for system redundancy. Deletion of the less conservative condition is more conservative by definition. Maintaining the system in a more conservative condition cannot create new challenges to components and systems that could adversely affect their ability to mitigate accident consequences or diminish the integrity of any fission product barrier. Therefore, the deletion of the LCO does not increase the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   Maintaining the system in a more conservative condition does not adversely affect any fission product barrier, nor does it alter the safety function of safety related systems, structures, and components depended upon for accident prevention or mitigation. Equipment important to safety will continue to function at its design capacity. No new equipment is being added, replaced, or taken away by the deletion of the LCO. Therefore, the possibility of a new or different kind of accident is not created.

3. Involve a significant reduction in the margin of safety.

   Safety analysis acceptance criteria continue to be satisfied for containment heat removal with deletion of this LCO. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Therefore, there are no significant hazards associated with the containment cooling systems change.

3.5 Proposed Condensate Storage Tank (CST) Changes

Description of CST Changes

Paragraph TS 3.4.c.1, Condensate Storage Tank, has been revised to increase the minimum volume from 39,000 gallons to 41,500 gallons. "Usable volume" has been added to the specification for clarification.

Paragraph TS 3.4.c.2, Condensate Storage Tank, has been revised to increase the minimum volume from 39,000 gallons to 41,500 gallons. "Usable volume" has been added to the specification for clarification.

Significant Hazards Determination for Proposed CST Changes

The basis of these changes is the station blackout (SBO) four-hour coping period. The stretch power uprate evaluations determined that the increased decay heat resulting from the increased core power required more CST inventory during the four-hour coping period. The uprate project evaluations associated with these changes demonstrate continued safe operation of the plant. Based on these evaluations, the following conclusions can be reached with respect to 10 CFR 50.92. The proposed changes will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The stretch power uprate project evaluations performed included a review of the SBO event. Results of the evaluation verified that with the increase in the CST inventory, the evaluation criteria continue to be met. Systems will continue to function as designed and system performance criteria will continue to be met. Additionally, dose consequences have been evaluated for the power uprate and results remain within the appropriate acceptance criteria. Therefore, the changes to CST inventory do not increase the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The changes have no adverse effect on any safety related system and do not change the performance or integrity of any safety related system. Additionally, no new safety related equipment is being added or changed as a result of the proposed changes in inventory. Therefore, the possibility of a new or different kind of accident is not created.

3. Involve a significant reduction in the margin of safety.

   The results of the SBO event review have verified that the analysis criteria continue to be met. Systems will continue to function as designed and system performance criteria continue to be met. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Therefore, there are no significant hazards associated with the changes in CST inventory.

3.6 Proposed Auxiliary Feedwater System Changes

Description of Auxiliary Feedwater System Changes

Section 3.4.b regarding the auxiliary feedwater (AFW) system has been revised to require three operable AFW trains prior to increasing reactor power above 1673 MWt. AFW trains are defined in the current TS basis section 3.4.b, page TS B3.4-1. The original TS requirements within the section have been reordered to accommodate the new TS requirement as follows:

Paragraph TS 3.4.b.2 has been renumbered as 3.4.b.4.

New paragraph TS 3.4.b.3 has been inserted stating the following: "The reactor power shall not be increased above 1673 MWt unless three trains of AFW are OPERABLE. If two of three AFW trains are inoperable, then within two hours, action shall be taken to reduce reactor power to ≤ 1673 MWt."

Paragraph TS 3.4.b.3 has been renumbered as 3.4.b.5 and the reference to TS 3.4.b.2 in the last sentence has been revised to reference TS 3.4.b.3 and TS 3.4.b.4.

Paragraph TS 3.4.b.4 has been renumbered as 3.4.b.2.

Paragraph TS 3.4.b.5 has been renumbered as 3.4.b.6 and reference to TS 3.4.b.2 replaced with TS 3.4.b.4.

Paragraph TS 3.4.b.6 has been renumbered as 3.4.b.7.

Significant Hazards Determination for Proposed Auxiliary Feedwater System Changes

Current analysis of the loss of normal feedwater (LONF) accident at power levels of 1673 MWt or less demonstrated that one AFW train at minimum flow capability provides sufficient heat removal capacity to mitigate the LONF accident. The new TS will require two AFW trains to be operable for power levels 1673 MWt or less to accommodate required flow for the accident analysis and the limiting single failure criteria of one AFW pump. Analysis of the LONF accident at an uprated power level of 1772 MWt demonstrated that two AFW pump trains at minimum flow capability provide sufficient heat removal capacity to mitigate the LONF accident at powers from 1673 MWt to 1772 MWt. The new TS will require three AFW trains be operable at power levels above 1673 MWt to meet the accident analysis AFW flow assumption and the limiting single failure of one AFW pump. The changes to AFW operability assure accident analysis and limiting single failure criteria are met. The proposed changes will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The LONF accident analyses have demonstrated that the TS required AFW trains at the minimum assumed flow capability provide sufficient heat removal capacity to mitigate the LONF accident such that acceptance criteria are satisfied. Single failure criteria are still met, and no physical system changes have been made. Dose consequences have been evaluated for the power uprate and the results remain within the appropriate acceptance criteria. Therefore, the changes to the auxiliary feedwater system technical specifications do not increase the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The change has no adverse effect on any safety related system and does not change the performance or integrity of any safety related system. Additionally, no new safety related equipment is being added or changed as a result of these proposed changes to technical specifications. Therefore, the possibility of a new or different kind of accident is not created.

3. Involve a significant reduction in the margin of safety.

   The LONF analysis supporting the proposed changes to technical specifications meets the appropriate acceptance criteria. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Therefore, there are no significant hazards associated with the auxiliary feedwater system technical specification changes.

3.7 Proposed Editorial and Administrative Changes

Description of Editorial and Administrative Changes

Paragraph TS 2.3.a.3.A, for f(ΔI), has been revised to change "An even function" to "A function".

Paragraph TS 2.3.a.3.A, for f(ΔI), has been revised to change f(ΔI) to f_1(ΔI). Paragraph TS 2.3.a.3.B, for f(ΔI), has been revised to change f(ΔI) to f_2(ΔI).

Revise Table TS 4.1-2, "Minimum Frequencies for Sampling Tests." Change the units in the frequency column for sampling test 7, "Secondary Coolant, b. Iodine Concentration," from 0.1 µCi/cc to 0.1 µCi/gram.

Significant Hazards Determination for Editorial and Administrative Changes

The changes to TS 2.3.a.3.A, for F(ΔI), are editorial. The changes are being made to make the TS consistent with the terminology and function notation used in NUREG-1431 (reference 8).

The change to Table TS 4.1-2 is also editorial as it changes the units for secondary coolant iodine concentration. This change in units makes the table value consistent with the TS 3.4.d for secondary limits, which changed the units during the fuel transition license amendment (reference 7).

These editorial and administrative changes have been reviewed with respect to the criteria of 10 CFR 50.92 and the conclusions below have been reached. The proposed changes will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The editorial and administrative changes do not affect the analysis performed in support of the stretch power uprate. Therefore, the changes do not increase the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The editorial and administrative changes do not affect the analysis performed in support of the stretch power uprate. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed editorial and administrative changes. Therefore, the possibility of a new or different kind of accident is not created.

3. Involve a significant reduction in the margin of safety.

   The editorial and administrative changes do not affect the analysis performed in support of the stretch power uprate. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Therefore, there are no significant hazards associated with the editorial changes.

4.0 Significant Hazards Determination Conclusion

In accordance with the requirements of 10 CFR 50.90, Nuclear Management Company (NMC) (licensee) hereby requests amendments to facility operating license DPR-43, for the Kewaunee Nuclear Power Plant. The purpose of the proposed amendments is to revise the operating license and the Technical Specifications to allow operation at an increased rated power of 1772 MWt.

Nuclear Management Company has evaluated the proposed amendments in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of the Kewaunee Nuclear Power Plant in accordance with the proposed amendments presents no significant hazards. The NMC evaluation against each of the criteria in 10 CFR 50.92 was performed in section 3.0 of this attachment following the appropriate proposed changes.

A comprehensive review of accident analysis, component and systems analysis, and radiological dose consequences was performed for the stretch power uprate. Analyses met the appropriate acceptance criteria, as explained in the earlier no significant hazards determinations. Therefore, operation of the KNPP in accordance with the proposed amendments will not result in a significant increase in the probability or consequences of any accident previously analyzed; will not result in a new or different kind of accident from any accident previously analyzed; and will not result in a significant reduction in a margin of safety. Therefore, operation of KNPP in accordance with the proposed amendments does not involve a significant hazards consideration.

5.0 Statement of Environmental Considerations

An environmental review summary is contained in attachment 4, report section 8.10, "Environmental Assessment." The conclusion of this review is that the proposed amendments do not involve a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22 (b), an environmental assessment of the proposed change is not required.

6.0 References

1. Letter NRC-02-024 from Mark E. Warner to Document Control Desk, "Revision to the Design Basis Radiological Analysis Accident Source Term," dated March 19, 2002 (TAC No. MB4596).

2. Letter NRC-02-067 from Mark E. Warner to Document Control Desk, "License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications Changes for Use of Westinghouse VANTAGE+ Fuel," dated July 26, 2002 (TAC No. MB5718).

3. Letter NRC-03-004 from Thomas Coutu to Document Control Desk, "License Amendment Request 193, Measurement Uncertainty Recapture Power Uprate for Kewaunee Nuclear Power Plant," dated January 13, 2003 (TAC No MB 7225).

4. Kewaunee Nuclear Power Plant – Issuance of Amendment Regarding Implementation of Alternate Source Term (TAC No. MB 4596), Amendment No. 166, dated March 17, 2003.

5. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000.

6. Letter NRC-03-016 from Thomas Coutu to Document Control Desk, "NMC Responses to NRC Request for Additional Information Concerning License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications," dated February 27, 2003 (TAC No MB 5718).

7. Kewaunee Nuclear Power Plant – Issuance of Amendment (TAC No. MB 5718) (Amendment No. 167, Regarding Use of Westinghouse 422 VANTAGE + Nuclear Fuel with PERFORMANCE + Features) with Safety Evaluation Report, dated April 4, 2003.

8. NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Volume 1, Revision 1, April 1995.

9. Letter NRC-03-032 from Thomas Coutu to Document Control Desk, "License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications, Confirmation of Use of WCAP-12488, Addendum 2," dated March 21, 2003 (TAC No. MB5718).

10. Letter NRC-03-047 from Thomas Coutu to Document Control Desk, "Responses to Requests for Additional Information Regarding License Amendment Request 193, Measurement Uncertainty Recapture Power Uprate for Kewaunee Nuclear Power Plant," dated April 30, 2003.

11. Safety Evaluation of Addendum 1 to Topical Report (TR) WCAP-10125-P-A, "Extended Burnup Evaluation of Westinghouse Fuel," dated April 14, 2003 (TAC No. MB7484).

12. Letter NRC-02-064 from Mark E. Warner to Document Control Desk, "License Amendment Request 185 to the Kewaunee Nuclear Power Plant Technical Specifications, 'Core Operating Limits Report Implementation,'" dated July 26, 2002.

13. Letter NRC-02-108 from Thomas Coutu to the Document Control Desk, "Response to Request for Additional Information Related to Proposed Revision to the Kewaunee Nuclear Power Plant Technical Specifications LAR 185, 'Core Operating Limits Report,'" dated December 19, 2002.

ATTACHMENT 2

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

May 22, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

<u>License Amendment Request 195</u>

Strike-Out Pages for License, Technical Specifications, and Bases

License, Page 3
TS 1.0-4
TS 2.1-1
TS B2.1-1 and TS B2.1-2
TS 2.3-2 and TS 2.3-3
TS B2.3-2
TS 3.3-4
TS B3.3-3
TS 3.4-2 through TS 3.4-4
TS B3.4-2 through TS B3.4-4
TS B3.8-1 and TS B3.8-2
Table TS 3.5-1
TS B4.5-1
Table TS 4.1-2

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR, Chapter 1: (1) Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70, (2) is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and (3) is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The NMC is authorized to operate the facility at steady-state reactor core power levels not in excess of ~~1673~~1772 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~—~~__are hereby incorporated in the license. The NMC shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

The NMC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the KNPP Fire Plan, and as referenced in the Updated Safety Analysis Report, and as approved in the Safety Evaluation Reports, dated November 25, 1977, and December 12, 1978 (and supplement dated February 13, 1981) subject to the following provision:

The NMC may make changes to the approved Fire Protection Program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(4) Physical Protection

The NMC shall fully implement and maintain in effect all provisions of the Commission-approved "Kewaunee Nuclear Power Plant Security Manual," Rev. 1, approved by the NRC on December 15, 1989, the "Kewaunee Nuclear Power Plant Security Force Training and Qualification Manual," Rev. 7, approved by the NRC on November 17, 1987, and the "Kewaunee Nuclear Power Plant Security Contingency Plan," Rev. 1, approved by the NRC on September 1, 1983. These manuals include amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p).

(5) Fuel Burnup

The maximum rod average burnup for any rod shall be limited to 60 GWD/MTU until completion of an NRC environmental assessment supporting an increased limit.

j. <u>MODES</u>

MODE	REACTIVITY $\Delta k/k$	COOLANT TEMP T_{avg} °F	FISSION POWER %
REFUELING	$\leq$ -5%	$\leq$ 140	~0
COLD SHUTDOWN	$\leq$ -1%	$\leq$ 200	~0
INTERMEDIATE SHUTDOWN	(1)	> 200 < 540	~0
HOT SHUTDOWN	(1)	$\geq$ 540	~0
HOT STANDBY	< 0.25%	~T_{oper}	< 2
OPERATING	< 0.25%	~T_{oper}	$\geq$ 2
LOW POWER PHYSICS TESTING	(To be specified by specific tests)		
(1) Refer to the required SHUTDOWN MARGIN as specified in the Core Operating Limits Report.			

k. <u>REACTOR CRITICAL</u>

The reactor is said to be critical when the neutron chain reaction is self-sustaining.

l. <u>REFUELING OPERATION</u>

REFUELING OPERATION is any operation involving movement of reactor vessel internal components (those that could affect the reactivity of the core) within the containment when the vessel head is unbolted or removed.

m. <u>RATED POWER</u>

RATED POWER is the steady-state reactor core output of ~~1,673~~<u>1.772</u> MWt.

n. <u>REPORTABLE EVENT</u>

A REPORTABLE EVENT is defined as any of those conditions specified in 10 CFR 50.73.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS - REACTOR CORE

APPLICABILITY

Applies to the limiting combination of thermal power, Reactor Coolant System pressure and coolant temperature during the OPERATING and HOT STANDBY MODES.

OBJECTIVE

To maintain the integrity of the fuel cladding.

SPECIFICATION

a. The combination of RATED POWER level, coolant pressure, and coolant temperature shall not exceed the limits specified in the COLR. The SAFETY LIMIT is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

b. The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.14 for the HTP DNB correlation and 1.17 for the WRB-1 DNB correlation.

c. The peak fuel centerline temperature shall be maintained < ~~4700~~5080°F decreasing by 58°F per 10,000 MWD/MTU of burnup.

BASIS - Safety Limits-Reactor Core (TS 2.1)

The reactor core safety limits shall not be exceeded during steady state operation, normal operational transients, and anticipated operational occurrences. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of the reactor core safety limits prevent overheating of the fuel and cladding as well as possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all OPERATING conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters of RATED POWER, reactor coolant temperature and pressure have been related to DNB through a DNB correlation. The DNB correlation has been developed to predict the DNB heat flux and the location of the DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and ~~anticipated~~ Condition I and II transients is limited to the DNBR limit. This minimum DNBR corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all OPERATING conditions.

The SAFETY LIMIT curves as provided in the Core Operating Report Limits Report ~~which~~ show the ~~allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation) represent the~~ loci of points of thermal power, reactor coolant system average temperature, and reactor coolant system pressure for which ~~either the~~ the minimum DNBR is ~~equal to~~ not less than the safety analysis limit, that fuel centerline temperature remains below melting, ~~the DNBR limit or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNBR ratio reaches the DNBR limit and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is ensured is below these lines.~~ that the average enthalpy at the exit of the core is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within limits defined by the DNBR correlation. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the

DNBR ratio reaches the DNBR limit and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is ensured is below the safety limit curves.

The curves are based on the nuclear hot channel factor limits of as specified in the COLR.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in TS 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits as specified in the COLR ensure that the ~~DNBR is always greater at partial power than at full power~~ increase in peaking factor is more than offset by the decrease in power level.

The Reactor Control and PROTECTION SYSTEM is designed to prevent any anticipated combination of transient conditions that would result in a DNBR less than the DNBR limit.

Two departure from nucleate boiling ratio (DNBR) correlations are used in the generation and validation of the safety limit curves: the WRB-1 DNBR correlation and the high thermal performance (HTP) DNBR correlation. The WRB-1 correlation applies to the Westinghouse 422 V+ fuel. The HTP correlation applies to FRA-ANP fuel with HTP spacers. The DNBR correlations have been qualified and approved for application to Kewaunee. The DNB correlation limits are 1.14 for the HTP DNBR correlation, and 1.17 for the WRB-1 DNBR correlation.

3. Reactor Coolant Temperature

A. Overtemperature

$$\Delta T \leq \Delta T_0 \left[K_1 - K_2 (T - T') \frac{1 + \tau_1 s}{1 + \tau_2 s} + K_3 (P - P') - f_1(\Delta I) \right]$$

where

ΔT_0	=	Indicated ΔT at RATED POWER, %
T	=	Average temperature, °F
T'	≤	[*]°F
P	=	Pressurizer pressure, psig
P'	=	[*] psig
K_1	=	[*]
K_2	=	[*]
K_3	=	[*]
τ_1	=	[*] sec.
τ_2	=	[*] sec.

~~F~~$f_1(\Delta I)$ = ~~An even~~ A function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that:

1. For $q_t - q_b$ within [*], [*] %, $f_1(\Delta I) = 0$.

2. For each percent that the magnitude of $q_t - q_b$ exceeds [*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

3. For each percent that the magnitude of $q_t - q_b$ exceed -[*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

Note: [*] As specified in the COLR

B. Overpower

$$\Delta T \leq \Delta T_0 \left[K_4 - K_5 \frac{\tau_3 s}{\tau_3 s + 1} T - K_6 (T - T') - f_2 (\Delta I) \right]$$

where

ΔT_0 = Indicated ΔT at RATED POWER, %

T = Average Temperature, °F

T' ≤ [*]°F

K_4 ≤ [*]

K_5 ≥ [*] for increasing T; [*] for decreasing T

K_6 ≥ [*] for T > T'; [*] for T < T'

τ_3 = [*] sec.

~~F~~$f_2(\Delta I)$ = 0 for all ΔI

Note: [*] As specified in the COLR

4. Reactor Coolant Flow

A. Low reactor coolant flow per loop ≥ 90% of normal indicated flow as measured by elbow taps.

B. Reactor coolant pump motor breaker open

1. Low frequency setpoint ≥ 55.0 Hz

2. Low voltage setpoint ≥ 75% of normal voltage

5. Steam Generators

Low-low steam generator water level ≥ 5% of narrow range instrument span.

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding a value at which fuel pellet centerline melting would occur, and includes corrections for ~~axial power distribution,~~ change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.(2)

The overpower and overtemperature PROTECTION SYSTEM setpoints include the effects of fuel densification and clad flattening on core SAFETY LIMITS.(4)

Reactor Coolant Flow

The low-flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.(5)

The undervoltage and low frequency reactor trips provide additional protection against a decrease in flow. The undervoltage setting provides a direct reactor trip and a reactor coolant pump breaker trip. The undervoltage setting ensures a reactor trip signal will be generated before the low-flow trip setting is reached. The low frequency setting provides only a reactor coolant pump breaker trip.

Steam Generators

The low-low steam generator water level reactor trip ensures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting the Auxiliary Feedwater System. (6)

Reactor Trip Interlocks

Specified reactor trips are bypassed at low power where they are not required for protection and would otherwise interfere with normal operation. The prescribed setpoints above which these trips are made functional ensures their availability in the power range where needed. Confirmation that bypasses are automatically removed at the prescribed setpoints will be determined by periodic testing. The reactor trips related to loss of one or both reactor coolant pumps are unblocked at approximately 10% of power.

Table TS 3.5-1 lists the various parameters and their setpoints which initiate safety injection signals. A safety injection signal (SIS) also initiates a reactor trip signal. The periodic testing will verify that safety injection signals perform their intended function. Refer to the basis of Section 3.5 of these specifications for details of SIS signals.

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(4) WCAP-8092
(5) USAR Section 14.1.8
(6) USAR Section 14.1.10

c. Containment Cooling Systems

1. Containment Spray and Containment Fancoil Units

A. The reactor shall not be made critical unless the following conditions are satisfied, except for LOW POWER PHYSICS TESTS and except as provided by TS 3.3.c.1.A.3.

1. Two containment spray trains are OPERABLE with each train comprised of:

(i) ONE containment spray pump.

(ii) An OPERABLE flow path consisting of all valves and piping associated with the above train of components and required to function during accident conditions. This flow path shall be capable of taking suction from the Refueling Water Storage Tank and from the containment sump.

2. TWO trains of containment fancoil units are OPERABLE with two fancoil units in each train.

3. During power operation or recovery from inadvertent trip, any one of the following conditions of inoperability may exist during the time intervals specified. If OPERABILITY is not restored within the time specified, then within 1 hour action shall be initiated to:

- Achieve HOT STANDBY within the next 6 hours.
- Achieve HOT SHUTDOWN within the following 6 hours.
- Achieve COLD SHUTDOWN within an additional 36 hours.

(i) One containment fancoil unit train may be out of service for 7 days provided the opposite containment fancoil unit train remains OPERABLE.

(ii) One containment spray train may be out of service for 72 hours provided the opposite containment spray train remains OPERABLE.

~~(iii) Both containment fancoil unit trains may be out of service for 72 hours provided both containment spray trains remain OPERABLE.~~

(~~iv~~iii) The same containment fancoil unit and containment spray trains may be out of service for 72 hours provided their opposite containment fancoil unit and containment spray trains remain OPERABLE.

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 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.

2. When the Reactor Coolant System temperature is > 350°F, if three auxiliary feedwater trains are discovered to be inoperable, initiate immediate action to restore one auxiliary feedwater train to OPERABLE status and suspend all LIMITING CONDITIONS FOR OPERATION requiring MODE changes until one auxiliary feedwater train is restored to OPERABLE status.~~When the Reactor Coolant System temperature is > 350°F, any of the following conditions of inoperability may exist during the time interval specified:~~

   ~~A. One auxiliary feedwater train may be inoperable for 72 hours.~~

   ~~B. Two auxiliary feedwater trains may be inoperable for 4 hours.~~

   ~~C. One steam supply to the turbine-driven auxiliary feedwater pump may be inoperable for 7 days.~~

3. ~~When the Reactor Coolant System temperature is > 350°F, an auxiliary feedwater pump low discharge pressure trip channel may be inoperable for a period not to exceed 4 hours. If this time period is exceeded, the associated auxiliary feedwater train shall be declared inoperable and the OPERABILITY requirements of TS 3.4.b.2 applied.~~The reactor power shall not be increased above 1673 MWt unless three trains of AFW are OPERABLE. If two of the three AFW trains are inoperable, then within two hours, reduce reactor power to ≤ 1673 MWt.

4. When the Reactor Coolant System temperature is > 350°F, any of the following conditions of inoperability may exist during the time interval specified:

   A. One auxiliary feedwater train may be inoperable for 72 hours.

   B. Two auxiliary feedwater trains may be inoperable for 4 hours.

   C. One steam supply to the turbine-driven auxiliary feedwater pump may be inoperable for 7 days.

~~When the Reactor Coolant System temperature is > 350°F, if three auxiliary feedwater trains are discovered to be inoperable, initiate immediate action to restore one auxiliary feedwater train to OPERABLE status and suspend all LIMITING CONDITIONS FOR OPERATION requiring MODE changes until one auxiliary feedwater train is restored to OPERABLE status.~~

5. ~~If the OPERABILITY requirements of TS 3.4.b.2 above are not met within the times specified, then within 1 hour action shall be initiated to:~~

   - ~~Achieve HOT STANDBY within 6 hours~~
   - ~~Achieve HOT SHUTDOWN within the following 6 hours~~
   - ~~Achieve and maintain the Reactor Coolant System temperature < 350°F within an additional 12 hours.~~When the Reactor Coolant System temperature is > 350°F, an auxiliary feedwater pump low discharge pressure trip channel may be inoperable for a period not to exceed 4 hours. If this time period is exceeded, the associated auxiliary feedwater train shall be declared inoperable and the OPERABILITY requirements of TS 3.4.b.3 and TS 3.4.b.4 applied.

6. ~~When reactor power is < 15% of RATED POWER, any of the following conditions may exist without declaring the corresponding auxiliary feedwater train inoperable:~~

   ~~A. The auxiliary feedwater 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.~~

   ~~C. Valves AFW-10A and AFW-10B may be in the closed position.~~If the OPERABILITY requirements of TS 3.4.b.4 above are not met within the times specified, then within 1 hour action shall be initiated to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System temperature < 350°F within an additional 12 hours.

7. When reactor power is < 15% of RATED POWER, any of the following conditions may exist without declaring the corresponding auxiliary feedwater train inoperable:

   A. The auxiliary feedwater 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.

   C. Valves AFW-10A and AFW-10B may be in the closed position.

c. Condensate Storage Tank

1. The Reactor Coolant System shall not be heated > 350°F unless a minimum usable volume of ~~39,000~~41,500 gallons of water is available in the condensate storage tanks.

2. If the Reactor Coolant System temperature is > 350°F and a minimum usable volume of ~~39,000~~41,500 gallons of water is not available in the condensate storage tanks, reactor operation may continue for up to 48 hours.

3. If the time limit of TS 3.4.c.2 above cannot be met, within 1 hour initiate action to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System temperature < 350°F within an additional 12 hours.

d. Secondary Activity Limits

1. The Reactor Coolant System shall not be heated > 350°F unless the DOSE EQUIVALENT Iodine-131 activity on the secondary side of the steam generators is ≤ 0.1 μCi/gram.

2. When the Reactor Coolant System temperature is $> 350^{\circ}F$, the DOSE EQUIVALENT Iodine-131 activity on the secondary side of the steam generators may exceed 0.1 μCi/gram for up to 48 hours.

3. If the requirement of TS 3.4.d.2 cannot be met, then within 1 hour action shall be initiated to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System temperature $< 350^{\circ}F$ within an additional 12 hours.

Two analyses apply to the Loss of Normal Feedwater event:

1. Analysis of the Loss of Normal Feedwater (LONF) event at 1772 MWt.

2. Analysis of the Loss of Normal Feedwater event at 1673 MWt.

One AFW pump provides adequate capacity to mitigate the consequences of the LONF event at 1673 MWt. In the LONF event at 1772 MWt, any two of the three AFW pumps are necessary to provide adequate heat removal capacity.

In the unlikely event of a loss of off-site electrical power to the plant, continued capability of decay heat removal would be ensured by the availability of either the steam-driven AFW pump or one of the two motor-driven AFW pumps, and by steam discharge to the atmosphere through the main steam safety valves. Each motor-driven pump and turbine-driven AFW pump is normally aligned to both steam generators. Valves AFW-10A and AFW-10B are normally open. Any single AFW pump can supply sufficient feedwater for removal of decay heat from the reactor.

As the plant is cooled down, heated up, or operated in a low power condition, AFW flow will have to be adjusted to maintain an adequate water inventory in the steam generators. This can be accomplished by any one of the following:

1. Throttling the discharge valves on the motor-driven AFW pumps

2. Closing one or both of the cross-connect flow valves

3. Stopping the pumps

If the main feedwater pumps are not in operation at the time, valves AFW-2A and AFW-2B must be throttled or the control switches for the AFW pumps located in the control room will have to be placed in the "pull out" position to prevent their continued operation and overfill of the steam generators. The cross-connect flow valves may be closed to specifically direct AFW flow. Manual action to re-initiate flow after it has been isolated is considered acceptable based on analyses performed by WPSC and the Westinghouse Electric Corporation. These analyses conservatively assumed the plant was at 100% initial power and demonstrated that operators have at least 10 minutes to manually initiate AFW during any design basis accident with no steam generator dryout or core damage. The placing of the AFW control switches in the "pull out" position, the closing of one or both cross-connect valves, and the closing or throttling of valves AFW-2A and AFW-2B are limited to situations when reactor power is <15% of RATED POWER to provide further margin in the analysis.

During accident conditions, the AFW System provides three functions:

1. Prevents thermal cycling of the steam generator tubesheet upon loss of the main feedwater pump

2. Removes residual heat from the Reactor Coolant System until the temperature drops below 300-350°F and the RHR System is capable of providing the necessary heat sink

3. Maintains a head of water in the steam generator following a loss-of-coolant accident

Each AFW pump provides 100% of the required capacity to the steam generators as assumed in the accident analyses performed at 1772 MWt to fulfill the above functions. ~~Since the AFW System is a safety features system, the backup pump is provided. This redundant motor-driven capability is also supplemented by the turbine-driven pump.~~ The exception is the LONF accident analysis performed at 1772 MWt. Based on the LONF accident analysis at 1772 MWt, two AFW pumps are required to provide adequate capacity.

The pumps are capable of automatic starting and can deliver full AFW flow within one minute after the signal for pump actuation. However, analyses from full power demonstrate that initiation of flow can be delayed for at least 10 minutes with no steam generator dryout or core damage. The head generated by the AFW pumps is sufficient to ensure that feedwater can be pumped into the steam generators when the safety valves are discharging and the supply source is at its lowest head.

Analyses by WPSC and the Westinghouse Electric Corporation show that AFW-2A and AFW-2B may be in the throttled or closed position, or the AFW pump control switches located in the control room may be in the "pull out" position without a compromise to safety. This does not constitute a condition of inoperability as listed in TS 3.4.b.1 or TS 3.4.b.~~2~~4. The analysis shows that diverse automatic reactor trips ensure a plant trip before any core damage or system overpressure occurs and that at least 10 minutes are available for the operators to manually initiate auxiliary feedwater flow (start AFW pumps or fully open AFW-2A and AFW-2B) for any credible accident from an initial power of 100%.

The OPERABILITY of the AFW System following a main steam line break (MSLB) was reviewed in our response to IE Bulletin 80-04. As a result of this review, requirements for the turbine-driven AFW pump were added to the Technical Specifications. In a secondary line break, it is assumed that the pump discharging to the intact steam generator fails and that the flow from the redundant motor-driven AFW pump is discharging out the break. Therefore, to meet single failure criteria, the turbine-driven AFW pump was added to Technical Specifications.

The OPERABILITY of the AFW system following a LONF event was analyzed as part of the stretch uprate. As a result of the analysis at 1772 MWt, requirements for three OPERABLE AFW trains prior to increasing power above 1673 MWt were added to the Technical Specifications. In a LONF event, it is assumed that one of the AFW pumps fails. Therefore, to meet single failure criteria, all three pumps are required to be OPERABLE prior to increasing power level above 1673 MWt.

For all ~~other~~ design basis accidents other than MSLB and the LONF at 1772 MWt, the two motor-driven AFW pumps supply sufficient redundancy to meet single failure criteria. ~~In a secondary line break, it is assumed that the pump discharging to the intact steam generator fails and that the flow from the redundant motor-driven AFW pump is discharging out the break. Therefore, to meet single failure criteria, the turbine-driven AFW pump was added to Technical Specifications.~~

The cross-connect valves (AFW-10A and AFW-10B) are normally maintained in the open position This provides an added degree of redundancy above what is required for all accidents except for a MSLB. During a MSLB, one of the cross-connect valves will have to be repositioned regardless if the valves are normally opened or closed. Therefore, the position of the cross-connect valves does not affect the performance of the turbine-driven AFW train. However, performance of the train is dependent on the ability of the valves to reposition. Although analyses have demonstrated that operation with the cross-connect valves closed is acceptable, the TS restrict operation with the valves closed to <15% of RATED POWER. At > 15% RATED POWER, closure of the cross-connect valves renders the TDAFW train inoperable.

An AFW train is defined as the AFW system piping, valves and pumps directly associated with providing AFW from the AFW pumps to the steam generators. The action with three trains inoperable is to maintain the plant in an OPERATING condition in which the AFW System is not needed for heat removal. When one train is restored, then the LIMITING CONDITIONS FOR OPERATION specified in TS 3.4.b.2, TS 3.4.b.3, and TS 3.4.b.4 are applied. The two and four hour clocks in TS 3.4.b.3 and TS 3.4.b.4 are started simultaneously. The two hour clock of TS 3.4.b.3 is for the power level restriction. The four hour clock of TS 3.4.b.4 is for starting the shutdown sequence. Should the plant shutdown be initiated with no AFW trains available, there would be no feedwater to the steam generators to cool the plant to 350°F when the RHR System could be placed into operation.

It is acceptable to exceed 350°F with an inoperable turbine-driven AFW train. However, OPERABILITY of the train must be demonstrated within 72 hours after exceeding 350°F or a plant shutdown must be initiated. This provides 72 hours with steam pressure for post-maintenance testing of the turbine AFW pump.

Condensate Storage Tank (TS 3.4.c)

The specified minimum usable water supply in the condensate storage tanks (CST) is sufficient for four hours of decay heat removal. The four hours are based on the Kewaunee site specific station blackout (loss of all AC power) coping duration requirement. Total CST water supply is maintained above a level that includes minimum usable water supply in technical specifications based on the station blackout analysis, allowance for flow to the condenser before isolation, allowance for AFW pump cooling, unusable level, and instrument error in each tank's level instrument.

The shutdown sequence of TS 3.4.c.3 allows for a safe and orderly shutdown of the reactor plant if the specified limits cannot be met. [1]

Secondary Activity Limits (TS 3.4.d)

The maximum dose that an individual may receive following an accident is specified in GDC 19 and 10 CFR 50.67. The limits on secondary coolant activity ensure that the calculated doses are held to the limits specified in GDC 19 and to a fraction of the 10 CFR 50.67 limits.

The secondary side of the steam generator's activity is limited to $\leq$ 0.1 μCi/gram DOSE EQUIVALENT I-131 to ensure the dose does not exceed the GDC-19 and 10 CFR 50.67 guidelines. The applicable accidents identified in the USAR[2] are analyzed assuming various inputs including steam generator activity of 0.1 μCi/gram DOSE EQUIVALENT I-131. The results obtained from these analyses indicate that the control room and off-site doses are within the acceptance criteria of GDC-19 and a fraction of 10 CFR 50.67 limits.

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(1) USAR Section 8.2.4
(2) USAR Section 14.0

BASIS – Refueling Operations (TS 3.8)

The equipment and general procedures to be utilized during REFUELING OPERATIONS are discussed in the USAR. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident occurs during the REFUELING OPERATIONS that would result in a hazard to public health and safety.[1] Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels (TS 3.8.a.2) and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

A minimum shutdown margin of greater than or equal to 5% $\Delta k/k$ must be maintained in the core. The boron concentration as specified in the COLR is sufficient to ensure an adequate margin of safety. The specification for REFUELING OPERATIONS shutdown margin is based on a dilution during refueling accident.[2] With an initial shutdown margin of 5% $\Delta k/k$, under the postulated accident conditions, it will take longer than 30 minutes for the reactor to go critical. This is ample time for the operator to recognize the audible high count rate signal, and isolate the reactor makeup water system. Periodic checks of refueling water boron concentration ensure that proper shutdown margin is maintained. Specification 3.8.a.6 allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

Interlocks are utilized during REFUELING OPERATIONS to ensure safe handling. Only one assembly at a time can be handled. The fuel handling hoist is dead weight tested prior to use to assure proper crane operation. It will not be possible to lift or carry heavy objects over the spent fuel pool when fuel is stored therein through interlocks and administrative procedures. Placement of additional spent fuel racks will be controlled by detailed procedures to prevent traverse directly above spent fuel.

The one hundred forty-eight hour decay time following plant shutdown ~~is consistent with the spent fuel pool cooling analysis and also~~ bounds the assumption used in the dose calculation for the fuel handling accident. A cycle-specific cooling analysis will be performed to verify that the spent fuel pool cooling system can maintain the pool temperature within allowable limits based on the one hundred forty-eight hour decay time. In the unlikely event that the analysis determines this time is not sufficient to maintain acceptable pool temperature, the analysis will determine the additional in core hold time required. The requirement for the spent fuel pool sweep system, including charcoal adsorbers, to be operating when spent fuel movement is being made provides added assurance that the off-site doses will be within acceptable limits in the event of a fuel handling accident. The spent fuel pool sweep system is designed to sweep the atmosphere above the refueling pool and release to the Auxiliary Building vent during fuel handling operations. Normally, the charcoal adsorbers are bypassed but for purification operation, the bypass dampers are closed routing the air flow through the charcoal adsorbers. If the dampers do not close tightly, bypass leakage could exist to negate the usefulness of the charcoal adsorber. If the spent fuel pool sweep system is found not to be operating, fuel handling within the Auxiliary Building will be terminated until the system can be restored to the operating condition.

The bypass dampers are integral to the filter housing. The test of the bypass leakage around the charcoal adsorbers will include the leakage through these dampers.

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[1] USAR Section 9.5.2
[2] USAR Section 14.1

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential radioiodine releases to the atmosphere. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP, respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency under test conditions which are more severe than accident conditions.

Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR Part ~~100~~50.67 for the accidents analyzed.

The spent fuel pool sweep system will be operated for the first month after reactor is shutdown for refueling during fuel handling and crane operations with loads over the pool. The potential consequences of a postulated fuel handling accident without the system are a very small fraction of the guidelines of 10 CFR Part ~~100~~ 50.67 after one month decay of the spent fuel. Heavy loads greater than one fuel assembly are not allowed over the spent fuel.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

A fuel handling accident in containment does not cause containment pressurization. One containment door in each personnel air lock can be closed following containment personnel evacuation and the containment ventilation and purge system has the capability to initiate automatic containment ventilation isolation to terminate a release path to the atmosphere.

The presence of a licensed senior reactor operator at the site and designated in charge provides qualified supervision of the REFUELING OPERATIONS during changes in core geometry.

Accident analysis assumes a charcoal adsorber efficiency of 90%.(3) To ensure the charcoal adsorbers maintain that efficiency throughout the operating cycle, a safety factor of 2 is used. Therefore, if accident analysis assumes a charcoal adsorber efficiency of 90%, this equates to a methyl iodide penetration of 10%. If a safety factor of 2 is assumed, the methyl iodide penetration is reduced to 5%. Thus, the acceptance criteria of 95% efficient will be used for the charcoal adsorbers.

Although committing to ASTM D3803-89, it was recognized that ASTM D3803-89 Standard references Military Standards MIL-F-51068D, Filter, Particulate High Efficiency, Fire Resistant, and MIL-F-51079A, Filter, Medium Fire Resistant, High Efficiency. These specifications have been revised and the latest revisions are, MIL-F-51068F and MIL-F-51079D. These revisions have been canceled and superseded by ASME AG-1, Code on Nuclear Air and Gas Treatment. ASME AG-1 is an acceptable substitution. Consequently, other referenced standards can be substituted if the new standard or methodology is shown to provide equivalent or superior performance to those referenced in ASTM D3803-89.

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(3) USAR TABLE 14.3-8, "Major Assumptions for Design Basis LOCA Analysis"

TABLE TS 3.5-1

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
1	High Containment Pressure (Hi)	Safety injection[1]	≤ 4 psig
2	High Containment Pressure (Hi-Hi)	a. Containment spray b. Steam line isolation of both lines	≤ 23 psig ≤ 17 psig
3	Pressurizer Low Pressure	Safety injection[1]	≥ 1815 psig
4	Low Steam Line Pressure	Safety injection[1] Lead time constant Lag time constant	≥ 500 psig ≥ 12 seconds ≤ 2 seconds
5	High Steam Flow in a Steam Line Coincident with Safety Injection and "Lo-Lo" T_{avg}	Steam line isolation of affected line[2]	≤ d/p corresponding to 0.745×10^6 lb/hr at 1005 psig ≥ 540°F
6	High-High Steam Flow in a Steam Line Coincident with Safety Injection	Steam line isolation of affected line[2]	≤ d/p corresponding to 4.~~5~~ 4 x 10^6 lb/hr at 735 psig
7	Forebay Level	Trip circ. water pumps	

[1] Initiates containment isolation, feedwater line isolation, shield building ventilation, auxiliary building special vent, and starting of all containment fans. In addition, the signal overrides any bypass on the accumulator valves.

[2] Confirm main steam isolation valves closure within 5 seconds when tested. d/p = differential pressure

~~Amendment No. 107~~LAR 195
~~04/07/94~~05/22/2003

BASIS

System Tests (TS 4.5.a)

The Safety Injection System and the Containment Vessel Internal Spray System are principal plant safety systems that are normally in standby during reactor operation. Complete systems tests cannot be performed when the reactor is OPERATING because a safety injection signal causes containment isolation, and a Containment Vessel Internal Spray System test requires the system to be temporarily disabled. The method of assuring OPERABILITY of these systems is therefore to combine system tests to be performed during periodic shutdowns with more frequent component tests, which can be performed during reactor operation.

The systems tests demonstrate proper automatic operation of the Safety Injection and Containment Vessel Internal Spray Systems. A test signal is applied to initiate automatic action, resulting in verification that the components received the safety injection signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.(1)

The Internal Containment Spray (ICS) System is designed to provide containment cooling in the event of a loss-of-coolant accident or steam line break accident, thereby ensuring the containment pressure does not exceed its design value of 46 psig at 268°F (100% R.H.).(2) ~~To ensure adequate cooling is available, calculations were performed to determine the ICS flow rate necessary to provide post-accident cooling. These calculations showed that a flow rate of 1300 gpm provides the required cooling capabilities for one train.~~ With the KNPP ICS system design, 76 properly functioning spray nozzles per train will adequately provide the required ICS flow rate ~~of 1300 gpm per train~~ for post accident cooling.

Component Tests - Containment Fancoil Units (TS 4.5.a.3)

Testing of the containment fancoil unit emergency discharge and backdraft dampers is performed to assure the integrity of the duct work post-LOCA.

Component Tests - Pumps (TS 4.5.b.1)

During reactor operation, the instrumentation which is depended upon to initiate safety injection and containment spray is checked daily and the initiating logic circuits are tested monthly (in accordance with TS 4.1). In addition, the active components (pumps and valves) are to be tested quarterly to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The quarterly test interval is based on the judgment that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), yet more frequent testing would result in increased wear over a long period of time.

(1) USAR Section 6.2
(2) USAR Section 6.4

TABLE TS 4.1-2

MINIMUM FREQUENCIES FOR SAMPLING TESTS

SAMPLING TESTS	TEST	FREQUENCY
3. Refueling Water Storage Tank Water Sample[7]	Boron Concentration	Monthly[8]
4. Deleted		
5. Accumulator	Boron Concentration	Monthly
6. Spent Fuel Pool	Boron Concentration	Monthly[9]
7. Secondary Coolant	a. Gross Beta or Gamma Activity b. Iodine Concentration	Weekly Weekly when gross beta or gamma activity $\geq$ 0.1 μCi/~~cc~~gram

[7] A refueling water storage tank (RWST) boron concentration sample does not have to be taken when the RWST is empty during REFUELING outages.
[8] And after adjusting tank contents.
[9] Sample will be taken monthly when fuel is in the pool.

~~Amendment No. 140~~LAR 195
~~10/27/98~~05/22/2003

ATTACHMENT 3

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

May 22, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 195

Revised Pages for License, Technical Specifications, and Bases

License, Page 3
TS 1.0-4
TS 2.1-1
TS B2.1-1 and TS B2.1-2
TS 2.3-2 and TS 2.3-3
TS B2.3-2
TS 3.3-4
TS B3.3-3
TS 3.4-2 and TS 3.4-3
TS B3.4-2 through TS B3.4-4
TS B3.8-1 and TS B3.8-2
Table TS 3.5-1
TS B4.5-1
Table TS 4.1-2

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR, Chapter 1: (1) Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70, (2) is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and (3) is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The NMC is authorized to operate the facility at steady-state reactor core power levels not in excess of 1772 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. are hereby incorporated in the license. The NMC shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

The NMC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the KNPP Fire Plan, and as referenced in the Updated Safety Analysis Report, and as approved in the Safety Evaluation Reports, dated November 25, 1977, and December 12, 1978 (and supplement dated February 13, 1981) subject to the following provision:

The NMC may make changes to the approved Fire Protection Program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(4) Physical Protection

The NMC shall fully implement and maintain in effect all provisions of the Commission-approved "Kewaunee Nuclear Power Plant Security Manual," Rev. 1, approved by the NRC on December 15, 1989, the "Kewaunee Nuclear Power Plant Security Force Training and Qualification Manual," Rev. 7, approved by the NRC on November 17, 1987, and the "Kewaunee Nuclear Power Plant Security Contingency Plan," Rev. 1, approved by the NRC on September 1, 1983. These manuals include amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p).

(5) Fuel Burnup

The maximum rod average burnup for any rod shall be limited to 60 GWD/MTU until completion of an NRC environmental assessment supporting an increased limit.

j. MODES

MODE	REACTIVITY Δk/k	COOLANT TEMP T_{avg} °F	FISSION POWER %
REFUELING	≤ -5%	≤ 140	~0
COLD SHUTDOWN	≤ -1%	≤ 200	~0
INTERMEDIATE SHUTDOWN	(1)	> 200 < 540	~0
HOT SHUTDOWN	(1)	≥ 540	~0
HOT STANDBY	< 0.25%	~T_{oper}	< 2
OPERATING	< 0.25%	~T_{oper}	≥ 2
LOW POWER PHYSICS TESTING	(To be specified by specific tests)		
(1) Refer to the required SHUTDOWN MARGIN as specified in the Core Operating Limits Report.			

k. REACTOR CRITICAL

The reactor is said to be critical when the neutron chain reaction is self-sustaining.

l. REFUELING OPERATION

REFUELING OPERATION is any operation involving movement of reactor vessel internal components (those that could affect the reactivity of the core) within the containment when the vessel head is unbolted or removed.

m. RATED POWER

RATED POWER is the steady-state reactor core output of 1,772 MWt.

n. REPORTABLE EVENT

A REPORTABLE EVENT is defined as any of those conditions specified in 10 CFR 50.73.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS - REACTOR CORE

APPLICABILITY

Applies to the limiting combination of thermal power, Reactor Coolant System pressure and coolant temperature during the OPERATING and HOT STANDBY MODES.

OBJECTIVE

To maintain the integrity of the fuel cladding.

SPECIFICATION

a. The combination of RATED POWER level, coolant pressure, and coolant temperature shall not exceed the limits specified in the COLR. The SAFETY LIMIT is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

b. The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.14 for the HTP DNB correlation and 1.17 for the WRB-1 DNB correlation.

c. The peak fuel centerline temperature shall be maintained < 5080°F decreasing by 58°F per 10,000 MWD/MTU of burnup.

BASIS - Safety Limits-Reactor Core (TS 2.1)

The reactor core safety limits shall not be exceeded during steady state operation, normal operational transients, and anticipated operational occurrences. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of the reactor core safety limits prevent overheating of the fuel and cladding as well as possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all OPERATING conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters of RATED POWER, reactor coolant temperature and pressure have been related to DNB through a DNB correlation. The DNB correlation has been developed to predict the DNB heat flux and the location of the DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and Condition I and II transients is limited to the DNBR limit. This minimum DNBR corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all OPERATING conditions.

The SAFETY LIMIT curves as provided in the Core Operating Report Limits Report show the loci of points of thermal power, reactor coolant system average temperature, and reactor coolant system pressure for which the minimum DNBR is not less than the safety analysis limit, that fuel centerline temperature remains below melting, that the average enthalpy at the exit of the core is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within limits defined by the DNBR correlation. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNBR ratio reaches the DNBR limit and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is ensured is below the safety limit curves.

The curves are based on the nuclear hot channel factor limits of as specified in the COLR.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in TS 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits as specified in the COLR ensure that the increase in peaking factor is more than offset by the decrease in power level.

The Reactor Control and PROTECTION SYSTEM is designed to prevent any anticipated combination of transient conditions that would result in a DNBR less than the DNBR limit.

Two departure from nucleate boiling ratio (DNBR) correlations are used in the generation and validation of the safety limit curves: the WRB-1 DNBR correlation and the high thermal performance (HTP) DNBR correlation. The WRB-1 correlation applies to the Westinghouse 422 V+ fuel. The HTP correlation applies to FRA-ANP fuel with HTP spacers. The DNBR correlations have been qualified and approved for application to Kewaunee. The DNB correlation limits are 1.14 for the HTP DNBR correlation, and 1.17 for the WRB-1 DNBR correlation.

3. Reactor Coolant Temperature

A. Overtemperature

$$\Delta T \leq \Delta T_0 \left[K_1 - K_2 (T - T') \frac{1 + \tau_1 s}{1 + \tau_2 s} + K_3 (P - P') - f_1(\Delta I) \right]$$

where

ΔT_0	=	Indicated ΔT at RATED POWER, %
T	=	Average temperature, °F
T'	≤	[*]°F
P	=	Pressurizer pressure, psig
P'	=	[*] psig
K_1	=	[*]
K_2	=	[*]
K_3	=	[*]
τ_1	=	[*] sec.
τ_2	=	[*] sec.
$f_1(\Delta I)$	=	A function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that:

1. For $q_t - q_b$ within [*], [*] %, $f_1(\Delta I) = 0$.

2. For each percent that the magnitude of $q_t - q_b$ exceeds [*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

3. For each percent that the magnitude of $q_t - q_b$ exceed -[*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

Note: [*] As specified in the COLR

B. Overpower

$$\Delta T \leq \Delta T_0 \left[K_4 - K_5 \frac{\tau_3 s}{\tau_3 s + 1} T - K_6 (T - T') - f_2 (\Delta I) \right]$$

where

ΔT_0 = Indicated ΔT at RATED POWER, %

T = Average Temperature, °F

T' $\leq$ [*]°F

K_4 $\leq$ [*]

K_5 $\geq$ [*] for increasing T; [*] for decreasing T

K_6 $\geq$ [*] for T > T'; [*] for T < T'

τ_3 = [*] sec.

$f_2(\Delta I)$ = 0 for all ΔI

Note: [*] As specified in the COLR

4. Reactor Coolant Flow

   A. Low reactor coolant flow per loop $\geq$ 90% of normal indicated flow as measured by elbow taps.

   B. Reactor coolant pump motor breaker open

      1. Low frequency setpoint $\geq$ 55.0 Hz

      2. Low voltage setpoint $\geq$ 75% of normal voltage

5. Steam Generators

   Low-low steam generator water level $\geq$ 5% of narrow range instrument span.

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding a value at which fuel pellet centerline melting would occur, and includes corrections for change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.[(2)]

The overpower and overtemperature PROTECTION SYSTEM setpoints include the effects of fuel densification and clad flattening on core SAFETY LIMITS.[(4)]

Reactor Coolant Flow

The low-flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.[(5)]

The undervoltage and low frequency reactor trips provide additional protection against a decrease in flow. The undervoltage setting provides a direct reactor trip and a reactor coolant pump breaker trip. The undervoltage setting ensures a reactor trip signal will be generated before the low-flow trip setting is reached. The low frequency setting provides only a reactor coolant pump breaker trip.

Steam Generators

The low-low steam generator water level reactor trip ensures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting the Auxiliary Feedwater System.[(6)]

Reactor Trip Interlocks

Specified reactor trips are bypassed at low power where they are not required for protection and would otherwise interfere with normal operation. The prescribed setpoints above which these trips are made functional ensures their availability in the power range where needed. Confirmation that bypasses are automatically removed at the prescribed setpoints will be determined by periodic testing. The reactor trips related to loss of one or both reactor coolant pumps are unblocked at approximately 10% of power.

Table TS 3.5-1 lists the various parameters and their setpoints which initiate safety injection signals. A safety injection signal (SIS) also initiates a reactor trip signal. The periodic testing will verify that safety injection signals perform their intended function. Refer to the basis of Section 3.5 of these specifications for details of SIS signals.

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(4) WCAP-8092
(5) USAR Section 14.1.8
(6) USAR Section 14.1.10

c. Containment Cooling Systems

1. Containment Spray and Containment Fancoil Units

A. The reactor shall not be made critical unless the following conditions are satisfied, except for LOW POWER PHYSICS TESTS and except as provided by TS 3.3.c.1.A.3.

1. Two containment spray trains are OPERABLE with each train comprised of:

(i) ONE containment spray pump.

(ii) An OPERABLE flow path consisting of all valves and piping associated with the above train of components and required to function during accident conditions. This flow path shall be capable of taking suction from the Refueling Water Storage Tank and from the containment sump.

2. TWO trains of containment fancoil units are OPERABLE with two fancoil units in each train.

3. During power operation or recovery from inadvertent trip, any one of the following conditions of inoperability may exist during the time intervals specified. If OPERABILITY is not restored within the time specified, then within 1 hour action shall be initiated to:

- Achieve HOT STANDBY within the next 6 hours.
- Achieve HOT SHUTDOWN within the following 6 hours.
- Achieve COLD SHUTDOWN within an additional 36 hours.

(i) One containment fancoil unit train may be out of service for 7 days provided the opposite containment fancoil unit train remains OPERABLE.

(ii) One containment spray train may be out of service for 72 hours provided the opposite containment spray train remains OPERABLE.

(iii) The same containment fancoil unit and containment spray trains may be out of service for 72 hours provided their opposite containment fancoil unit and containment spray trains remain OPERABLE.

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 of the following combinations will provide sufficient cooling to limit containment pressure to less than design values: four fancoil units 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).

[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.

2. When the Reactor Coolant System temperature is > 350°F, if three auxiliary feedwater trains are discovered to be inoperable, initiate immediate action to restore one auxiliary feedwater train to OPERABLE status and suspend all LIMITING CONDITIONS FOR OPERATION requiring MODE changes until one auxiliary feedwater train is restored to OPERABLE status.

3. The reactor power shall not be increased above 1673 MWt unless three trains of AFW are OPERABLE. If two of the three AFW trains are inoperable, then within two hours, reduce reactor power to $\leq$ 1673 MWt.

4. When the Reactor Coolant System temperature is > 350°F, any of the following conditions of inoperability may exist during the time interval specified:

   A. One auxiliary feedwater train may be inoperable for 72 hours.

   B. Two auxiliary feedwater trains may be inoperable for 4 hours.

   C. One steam supply to the turbine-driven auxiliary feedwater pump may be inoperable for 7 days.

5. When the Reactor Coolant System temperature is > 350°F, an auxiliary feedwater pump low discharge pressure trip channel may be inoperable for a period not to exceed 4 hours. If this time period is exceeded, the associated auxiliary feedwater train shall be declared inoperable and the OPERABILITY requirements of TS 3.4.b.3 and TS 3.4.b.4 applied.

6. If the OPERABILITY requirements of TS 3.4.b.4 above are not met within the times specified, then within 1 hour action shall be initiated to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System temperature < 350°F within an additional 12 hours.

7. When reactor power is < 15% of RATED POWER, any of the following conditions may exist without declaring the corresponding auxiliary feedwater train inoperable:

   A. The auxiliary feedwater 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.

   C. Valves AFW-10A and AFW-10B may be in the closed position.

c. Condensate Storage Tank

1. The Reactor Coolant System shall not be heated > 350°F unless a minimum usable volume of 41,500 gallons of water is available in the condensate storage tanks.

2. If the Reactor Coolant System temperature is > 350°F and a minimum usable volume of 41,500 gallons of water is not available in the condensate storage tanks, reactor operation may continue for up to 48 hours.

3. If the time limit of TS 3.4.c.2 above cannot be met, within 1 hour initiate action to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System temperature < 350°F within an additional 12 hours.

d. Secondary Activity Limits

1. The Reactor Coolant System shall not be heated > 350°F unless the DOSE EQUIVALENT Iodine-131 activity on the secondary side of the steam generators is ≤ 0.1 μCi/gram.

2. When the Reactor Coolant System temperature is > 350°F, the DOSE EQUIVALENT Iodine-131 activity on the secondary side of the steam generators may exceed 0.1 μCi/gram for up to 48 hours.

3. If the requirement of TS 3.4.d.2 cannot be met, then within 1 hour action shall be initiated to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System temperature < 350°F within an additional 12 hours.

Two analyses apply to the Loss of Normal Feedwater event:

1. Analysis of the Loss of Normal Feedwater (LONF) event at 1772 MWt.

2. Analysis of the Loss of Normal Feedwater event at 1673 MWt.

One AFW pump provides adequate capacity to mitigate the consequences of the LONF event at 1673 MWt. In the LONF event at 1772 MWt, any two of the three AFW pumps are necessary to provide adequate heat removal capacity.

In the unlikely event of a loss of off-site electrical power to the plant, continued capability of decay heat removal would be ensured by the availability of either the steam-driven AFW pump or one of the two motor-driven AFW pumps, and by steam discharge to the atmosphere through the main steam safety valves. Each motor-driven pump and turbine-driven AFW pump is normally aligned to both steam generators. Valves AFW-10A and AFW-10B are normally open. Any single AFW pump can supply sufficient feedwater for removal of decay heat from the reactor.

As the plant is cooled down, heated up, or operated in a low power condition, AFW flow will have to be adjusted to maintain an adequate water inventory in the steam generators. This can be accomplished by any one of the following:

1. Throttling the discharge valves on the motor-driven AFW pumps

2. Closing one or both of the cross-connect flow valves

3. Stopping the pumps

If the main feedwater pumps are not in operation at the time, valves AFW-2A and AFW-2B must be throttled or the control switches for the AFW pumps located in the control room will have to be placed in the "pull out" position to prevent their continued operation and overfill of the steam generators. The cross-connect flow valves may be closed to specifically direct AFW flow. Manual action to re-initiate flow after it has been isolated is considered acceptable based on analyses performed by WPSC and the Westinghouse Electric Corporation. These analyses conservatively assumed the plant was at 100% initial power and demonstrated that operators have at least 10 minutes to manually initiate AFW during any design basis accident with no steam generator dryout or core damage. The placing of the AFW control switches in the "pull out" position, the closing of one or both cross-connect valves, and the closing or throttling of valves AFW-2A and AFW-2B are limited to situations when reactor power is <15% of RATED POWER to provide further margin in the analysis.

During accident conditions, the AFW System provides three functions:

1. Prevents thermal cycling of the steam generator tubesheet upon loss of the main feedwater pump

2. Removes residual heat from the Reactor Coolant System until the temperature drops below 300-350°F and the RHR System is capable of providing the necessary heat sink

3. Maintains a head of water in the steam generator following a loss-of-coolant accident

Each AFW pump provides 100% of the required capacity to the steam generators as assumed in the accident analyses performed at 1772 MWt to fulfill the above functions. The exception is the LONF accident analysis performed at 1772 MWt. Based on the LONF accident analysis at 1772 MWt, two AFW pumps are required to provide adequate capacity.

The pumps are capable of automatic starting and can deliver full AFW flow within one minute after the signal for pump actuation. However, analyses from full power demonstrate that initiation of flow can be delayed for at least 10 minutes with no steam generator dryout or core damage. The head generated by the AFW pumps is sufficient to ensure that feedwater can be pumped into the steam generators when the safety valves are discharging and the supply source is at its lowest head.

Analyses by WPSC and the Westinghouse Electric Corporation show that AFW-2A and AFW-2B may be in the throttled or closed position, or the AFW pump control switches located in the control room may be in the "pull out" position without a compromise to safety. This does not constitute a condition of inoperability as listed in TS 3.4.b.1 or TS 3.4.b.4. The analysis shows that diverse automatic reactor trips ensure a plant trip before any core damage or system overpressure occurs and that at least 10 minutes are available for the operators to manually initiate auxiliary feedwater flow (start AFW pumps or fully open AFW-2A and AFW-2B) for any credible accident from an initial power of 100%.

The OPERABILITY of the AFW System following a main steam line break (MSLB) was reviewed in our response to IE Bulletin 80-04. As a result of this review, requirements for the turbine-driven AFW pump were added to the Technical Specifications. In a secondary line break, it is assumed that the pump discharging to the intact steam generator fails and that the flow from the redundant motor-driven AFW pump is discharging out the break. Therefore, to meet single failure criteria, the turbine-driven AFW pump was added to Technical Specifications.

The OPERABILITY of the AFW system following a LONF event was analyzed as part of the stretch uprate. As a result of the analysis at 1772 MWt, requirements for three OPERABLE AFW trains prior to increasing power above 1673 MWt were added to the Technical Specifications. In a LONF event, it is assumed that one of the AFW pumps fails. Therefore, to meet single failure criteria, all three pumps are required to be OPERABLE prior to increasing power level above 1673 MWt.

For all design basis accidents other than MSLB and the LONF at 1772 MWt, the two motor-driven AFW pumps supply sufficient redundancy to meet single failure criteria.

The cross-connect valves (AFW-10A and AFW-10B) are normally maintained in the open position This provides an added degree of redundancy above what is required for all accidents except for a MSLB. During a MSLB, one of the cross-connect valves will have to be repositioned regardless if the valves are normally opened or closed. Therefore, the position of the cross-connect valves does not affect the performance of the turbine-driven AFW train. However, performance of the train is dependent on the ability of the valves to reposition. Although analyses have demonstrated that operation with the cross-connect valves closed is acceptable, the TS restrict operation with the valves closed to <15% of RATED POWER. At > 15% RATED POWER, closure of the cross-connect valves renders the TDAFW train inoperable.

An AFW train is defined as the AFW system piping, valves and pumps directly associated with providing AFW from the AFW pumps to the steam generators. The action with three trains inoperable is to maintain the plant in an OPERATING condition in which the AFW System is not needed for heat removal. When one train is restored, then the LIMITING CONDITIONS FOR OPERATION specified in TS 3.4.b.2, TS 3.4.b.3, and TS 3.4.b.4 are applied. The two and four hour clocks in TS 3.4.b.3 and TS 3.4.b.4 are started simultaneously. The two hour clock of TS 3.4.b.3 is for the power level restriction. The four hour clock of TS 3.4.b.4 is for starting the shutdown sequence. Should the plant shutdown be initiated with no AFW trains available, there would be no feedwater to the steam generators to cool the plant to 350°F when the RHR System could be placed into operation.

It is acceptable to exceed 350°F with an inoperable turbine-driven AFW train. However, OPERABILITY of the train must be demonstrated within 72 hours after exceeding 350°F or a plant shutdown must be initiated. This provides 72 hours with steam pressure for post-maintenance testing of the turbine AFW pump.

Condensate Storage Tank (TS 3.4.c)

The specified minimum usable water supply in the condensate storage tanks (CST) is sufficient for four hours of decay heat removal. The four hours are based on the Kewaunee site specific station blackout (loss of all AC power) coping duration requirement. Total CST water supply is maintained above a level that includes minimum usable water supply in technical specifications based on the station blackout analysis, allowance for flow to the condenser before isolation, allowance for AFW pump cooling, unusable level, and instrument error in each tank's level instrument.

The shutdown sequence of TS 3.4.c.3 allows for a safe and orderly shutdown of the reactor plant if the specified limits cannot be met. [(1)]

Secondary Activity Limits (TS 3.4.d)

The maximum dose that an individual may receive following an accident is specified in GDC 19 and 10 CFR 50.67. The limits on secondary coolant activity ensure that the calculated doses are held to the limits specified in GDC 19 and to a fraction of the 10 CFR 50.67 limits.

The secondary side of the steam generator's activity is limited to $\leq$ 0.1 μCi/gram DOSE EQUIVALENT I-131 to ensure the dose does not exceed the GDC-19 and 10 CFR 50.67 guidelines. The applicable accidents identified in the USAR[(2)] are analyzed assuming various inputs including steam generator activity of 0.1 μCi/gram DOSE EQUIVALENT I-131. The results obtained from these analyses indicate that the control room and off-site doses are within the acceptance criteria of GDC-19 and a fraction of 10 CFR 50.67 limits.

[(1)] USAR Section 8.2.4
[(2)] USAR Section 14.0

BASIS – Refueling Operations (TS 3.8)

The equipment and general procedures to be utilized during REFUELING OPERATIONS are discussed in the USAR. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident occurs during the REFUELING OPERATIONS that would result in a hazard to public health and safety.[1] Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels (TS 3.8.a.2) and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

A minimum shutdown margin of greater than or equal to 5% $\Delta k/k$ must be maintained in the core. The boron concentration as specified in the COLR is sufficient to ensure an adequate margin of safety. The specification for REFUELING OPERATIONS shutdown margin is based on a dilution during refueling accident.[2] With an initial shutdown margin of 5% $\Delta k/k$, under the postulated accident conditions, it will take longer than 30 minutes for the reactor to go critical. This is ample time for the operator to recognize the audible high count rate signal, and isolate the reactor makeup water system. Periodic checks of refueling water boron concentration ensure that proper shutdown margin is maintained. Specification 3.8.a.6 allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

Interlocks are utilized during REFUELING OPERATIONS to ensure safe handling. Only one assembly at a time can be handled. The fuel handling hoist is dead weight tested prior to use to assure proper crane operation. It will not be possible to lift or carry heavy objects over the spent fuel pool when fuel is stored therein through interlocks and administrative procedures. Placement of additional spent fuel racks will be controlled by detailed procedures to prevent traverse directly above spent fuel.

The one hundred forty-eight hour decay time following plant shutdown bounds the assumption used in the dose calculation for the fuel handling accident. A cycle-specific cooling analysis will be performed to verify that the spent fuel pool cooling system can maintain the pool temperature within allowable limits based on the one hundred forty-eight hour decay time. In the unlikely event that the analysis determines this time is not sufficient to maintain acceptable pool temperature, the analysis will determine the additional in core hold time required. The requirement for the spent fuel pool sweep system, including charcoal adsorbers, to be operating when spent fuel movement is being made provides added assurance that the off-site doses will be within acceptable limits in the event of a fuel handling accident. The spent fuel pool sweep system is designed to sweep the atmosphere above the refueling pool and release to the Auxiliary Building vent during fuel handling operations. Normally, the charcoal adsorbers are bypassed but for purification operation, the bypass dampers are closed routing the air flow through the charcoal adsorbers. If the dampers do not close tightly, bypass leakage could exist to negate the usefulness of the charcoal adsorber. If the spent fuel pool sweep system is found not to be operating, fuel handling within the Auxiliary Building will be terminated until the system can be restored to the operating condition.

The bypass dampers are integral to the filter housing. The test of the bypass leakage around the charcoal adsorbers will include the leakage through these dampers.

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(1) USAR Section 9.5.2
(2) USAR Section 14.1

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential radioiodine releases to the atmosphere. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP, respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency under test conditions which are more severe than accident conditions.

Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR Part 50.67 for the accidents analyzed.

The spent fuel pool sweep system will be operated for the first month after reactor is shutdown for refueling during fuel handling and crane operations with loads over the pool. The potential consequences of a postulated fuel handling accident without the system are a very small fraction of the guidelines of 10 CFR Part 50.67 after one month decay of the spent fuel. Heavy loads greater than one fuel assembly are not allowed over the spent fuel.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

A fuel handling accident in containment does not cause containment pressurization. One containment door in each personnel air lock can be closed following containment personnel evacuation and the containment ventilation and purge system has the capability to initiate automatic containment ventilation isolation to terminate a release path to the atmosphere.

The presence of a licensed senior reactor operator at the site and designated in charge provides qualified supervision of the REFUELING OPERATIONS during changes in core geometry.

Accident analysis assumes a charcoal adsorber efficiency of 90%.[(3)] To ensure the charcoal adsorbers maintain that efficiency throughout the operating cycle, a safety factor of 2 is used. Therefore, if accident analysis assumes a charcoal adsorber efficiency of 90%, this equates to a methyl iodide penetration of 10%. If a safety factor of 2 is assumed, the methyl iodide penetration is reduced to 5%. Thus, the acceptance criteria of 95% efficient will be used for the charcoal adsorbers.

Although committing to ASTM D3803-89, it was recognized that ASTM D3803-89 Standard references Military Standards MIL-F-51068D, Filter, Particulate High Efficiency, Fire Resistant, and MIL-F-51079A, Filter, Medium Fire Resistant, High Efficiency. These specifications have been revised and the latest revisions are, MIL-F-51068F and MIL-F-51079D. These revisions have been canceled and superseded by ASME AG-1, Code on Nuclear Air and Gas Treatment. ASME AG-1 is an acceptable substitution. Consequently, other referenced standards can be substituted if the new standard or methodology is shown to provide equivalent or superior performance to those referenced in ASTM D3803-89.

[(3)] USAR TABLE 14.3-8, "Major Assumptions for Design Basis LOCA Analysis"

TABLE TS 3.5-1

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
1	High Containment Pressure (Hi)	Safety injection[1]	$\leq$ 4 psig
2	High Containment Pressure (Hi-Hi)	a. Containment spray b. Steam line isolation of both lines	$\leq$ 23 psig $\leq$ 17 psig
3	Pressurizer Low Pressure	Safety injection[1]	$\geq$ 1815 psig
4	Low Steam Line Pressure	Safety injection[1] Lead time constant Lag time constant	$\geq$ 500 psig $\geq$ 12 seconds $\leq$ 2 seconds
5	High Steam Flow in a Steam Line Coincident with Safety Injection and "Lo-Lo" T_{avg}	Steam line isolation of affected line[2]	$\leq$ d/p corresponding to 0.745×10^6 lb/hr at 1005 psig $\geq$ 540°F
6	High-High Steam Flow in a Steam Line Coincident with Safety Injection	Steam line isolation of affected line[2]	$\leq$ d/p corresponding to 4.4×10^6 lb/hr at 735 psig
7	Forebay Level	Trip circ. water pumps	

[1] Initiates containment isolation, feedwater line isolation, shield building ventilation, auxiliary building special vent, and starting of all containment fans. In addition, the signal overrides any bypass on the accumulator valves.

[2] Confirm main steam isolation valves closure within 5 seconds when tested. d/p = differential pressure

BASIS

System Tests (TS 4.5.a)

The Safety Injection System and the Containment Vessel Internal Spray System are principal plant safety systems that are normally in standby during reactor operation. Complete system tests cannot be performed when the reactor is OPERATING because a safety injection signal causes containment isolation, and a Containment Vessel Internal Spray System test requires the system to be temporarily disabled. The method of assuring OPERABILITY of these systems is therefore to combine system tests to be performed during periodic shutdowns with more frequent component tests, which can be performed during reactor operation.

The system tests demonstrate proper automatic operation of the Safety Injection and Containment Vessel Internal Spray Systems. A test signal is applied to initiate automatic action, resulting in verification that the components received the safety injection signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.(1)

The Internal Containment Spray (ICS) System is designed to provide containment cooling in the event of a loss-of-coolant accident or steam line break accident, thereby ensuring the containment pressure does not exceed its design value of 46 psig at 268°F (100% R.H.).(2) With the KNPP ICS system design, 76 properly functioning spray nozzles per train will adequately provide the required ICS flow rate for post accident cooling.

Component Tests - Containment Fancoil Units (TS 4.5.a.3)

Testing of the containment fancoil unit emergency discharge and backdraft dampers is performed to assure the integrity of the duct work post-LOCA.

Component Tests - Pumps (TS 4.5.b.1)

During reactor operation, the instrumentation which is depended upon to initiate safety injection and containment spray is checked daily and the initiating logic circuits are tested monthly (in accordance with TS 4.1). In addition, the active components (pumps and valves) are to be tested quarterly to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The quarterly test interval is based on the judgment that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), yet more frequent testing would result in increased wear over a long period of time.

(1) USAR Section 6.2
(2) USAR Section 6.4

TABLE TS 4.1-2

MINIMUM FREQUENCIES FOR SAMPLING TESTS

SAMPLING TESTS	TEST	FREQUENCY
3. Refueling Water Storage Tank Water Sample[7]	Boron Concentration	Monthly[8]
4. Deleted		
5. Accumulator	Boron Concentration	Monthly
6. Spent Fuel Pool	Boron Concentration	Monthly[9]
7. Secondary Coolant	a. Gross Beta or Gamma Activity b. Iodine Concentration	Weekly Weekly when gross beta or gamma activity ≥ 0.1 μCi/gram

[7] A refueling water storage tank (RWST) boron concentration sample does not have to be taken when the RWST is empty during REFUELING outages.
[8] And after adjusting tank contents.
[9] Sample will be taken monthly when fuel is in the pool.

ATTACHMENT 4

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

May 22, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 195

WCAP-16040-P
"Power Uprate Project, Kewaunee Nuclear Power Plant, NSSS and BOP Licensing Report,"
dated February 2003 (Proprietary)

ATTACHMENT 5

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

May 22, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 195

WCAP-16040-NP
"Power Uprate Project, Kewaunee Nuclear Power Plant, NSSS and BOP Licensing Report"
dated February 2003 (Non-Proprietary)

ATTACHMENT 6

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

May 22, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 195

Westinghouse Authorization Letter, CAW-03-1603, and Accompanying Affidavit, Proprietary Information Notice, and Copyright Notice for Attachment 4

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
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USA

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

Direct tel: (412) 374-5282
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e-mail: Sepp1ha@westinghouse.com

Our ref: CAW-03-1603

February 27, 2003

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: "Kewaunee Nuclear Power Plant, Power Uprate Project, Volume 1, NSSS and BOP Licensing Report, WCAP-16040-P" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-03-1603 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-03-1603 and should be addressed to the undersigned.

Very truly yours,

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

Enclosures

cc: S. J. Collins
D. Holland/NRR
B. Benney/NRR

CAW-03-1603

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 19th day
of May, 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 "Kewaunee Nuclear Power Plant, Power Uprate Project, Volume 1, NSSS and BOP Licensing Report, WCAP-16040-P" (Proprietary), dated February 2003, 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 Power 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 licensing submitals.

(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 licensing submittals.

(b) Westinghouse can sell support and defense of the technology to its customers in the licensing process.

(c) 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).

COPYRIGHT NOTICE

ATTACHMENT 7

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

May 22, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 195

List of Regulatory Commitments

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by NMC in this document, 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: 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. 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.	 a. Complete. 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.