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Kewaunee Nuclear Power Plant
Operated by Nuclear Management Company, LLC

May 5, 2005

NRC-05-057
10 CFR 50.90

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 213 To The Kewaunee Nuclear Power Plant Technical Specifications: Auxiliary Feedwater Pump Protection

Pursuant to 10 CFR 50.90 Nuclear Management Company, LLC (NMC) requests Nuclear Regulatory Commission (NRC) expeditious review and approval of a proposed license amendment request (LAR) for the Kewaunee Nuclear Power Plant (KNPP). NMC proposes Technical Specification (TS) changes to modify the auxiliary feedwater (AFW) pump suction protection requirements. The proposed TS changes include 3.4.b, "Auxiliary Feedwater System" and TS Table 4.1-1, "Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels." NMC also proposes licensing basis changes to revise the functionality of the discharge pressure switches to provide pump runout protection, which requires operator actions to restore the AFW pumps for specific post-accident recovery activities. NMC has evaluated these proposed changes in accordance with 10 CFR 50.92 and concluded that they involve no significant hazards consideration.

Enclosure 1 provides a detailed description of the proposed change, background and technical analysis, No Significant Hazards Consideration Determination, and Environmental Review Consideration. Enclosure 2 provides the revised TS pages reflecting the proposed change. Enclosure 3 provides the annotated TS pages showing the changes proposed. Enclosures 4 and 5 provide marked up and annotated TS bases pages for information purposes.

NMC requests approval of the proposed amendment by June 6, 2005 to support restart of the Kewaunee Nuclear Power Plant. The license amendment will be implemented upon approval.

In accordance with 10 CFR 50.91, NMC is notifying the State of Wisconsin of this LAR by transmitting a copy of this letter and attachments to the designated state official.

Summary of Commitments

This letter contains one new commitment and no revisions to existing commitments.

- The operator actions to restore the AFW pumps for specific post-accident recovery activities will be time validated and provided to the NRC for review prior to exceeding 350°F in the reactor coolant system.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on May 5, 2005.

Craig W. Lambert
Site Vice President, Kewaunee Nuclear Power Plant
Nuclear Management Company, LLC

Enclosures (5)

cc: Administrator, Region III, USNRC
Senior Resident Inspector, Kewaunee, USNRC
Project Manager, Kewaunee, USNRC
Public Service Commission of Wisconsin

ENCLOSURE 1

**NUCLEAR MANAGEMENT COMPANY, LLC,
EVALUATION OF LICENSE AMENDMENT REQUEST 213 TO KEWAUNEE NUCLEAR POWER PLANT, OPERATING LICENSE NO. DPR-43, DOCKET NO. 50-305**

1.0 DESCRIPTION

This license amendment request (LAR) is to amend Operating License DPR-43 for the Kewaunee Nuclear Power Plant (KNPP).

The Nuclear Management Company, LLC (NMC) requests Nuclear Regulatory Commission (NRC) review and approval of the proposed changes to Technical Specifications (TS) 3.4.b, "Auxiliary Feedwater (AFW) System" and TS Table 4.1-1, "Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels." These proposed changes are required to ensure the AFW pumps are protected against loss of suction. NMC also requests NRC review and approval of the licensing basis changes to revise the functionality of the discharge pressure switches to provide pump runout protection, which requires operator actions to restore the AFW pumps for specific post-accident recovery activities.

2.0 PROPOSED CHANGE

A brief description of the associated proposed TS and TS Bases changes is provided below along with a discussion of the justification for each change. The specific wording changes to the TS and Bases are provided in Enclosures 2 through 5.

NMC is requesting the following changes be made to the KNPP TS for the AFW system.

1) TS 3.4.b.1: Add Limiting Conditions for Operation requirements for the new auxiliary feedwater pump suction pressure trip channels.

2) TS 3.4.b.5: Add required actions and completion time for an inoperable AFW pump low suction pressure trip. The clause, "or more than one pump's trip channel(s) are

inoperable," has been added to provide clarification that only a single pump's trip channel(s) are allowed inoperable for the four hour period. Multiple pumps' inoperable channel(s) require entry into the action of TS 3.4.b.3 and 3.4.b.4.

3) TS Table 4.1.1: Item 43, Change the test frequency from each refueling cycle to quarterly and add a remark that states that verification of relay setpoints is not required for the quarterly test.

4) TS Table 4.1.1: Item 46, Add new Item 46 for the AFW pump Low Suction Pressure Trip surveillances.

The KNPP TS Bases have also been modified to explain the protective function of the low suction and discharge pressure trip channels, and discuss required operator actions to restore the AFW pumps for specific post-accident recovery activities. The affected TS Bases pages are included for information.

In summary, these TS and licensing basis changes reflect modifications and evaluations which ensure the AFW pumps will perform their safety function when these changes are implemented.

3.0 BACKGROUND

Following the Three Mile Island Unit 2 accident in 1979, the NRC recommended licensees evaluate the design of their AFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. This recommendation was for plants that had unprotected normal AFW system water supplies, such as KNPP. In response to these recommendations, in 1983 Wisconsin Public Service Corporation (WPSC) (WPSC was the KNPP operating entity prior to 2001 when NMC became the operating entity) informed the NRC that an evaluation had been performed of the AFW system design and WPSC concluded that protection of the pumps was necessary following a loss of suction event. This protection was to have been provided by an automatic pump trip on low suction pressure.

In 1993, WPSC notified the NRC that a discharge pressure trip would be installed rather than a suction pressure trip because of sub-atmospheric conditions at the AFW pump suction. WPSC determined a low discharge pressure trip would best provide the desired net positive suction head (NPSH) protection. The NRC responded in June of 1993, acknowledging that a discharge pressure trip could be installed in lieu of a suction pressure trip and stating that the Staff had reviewed these changes in WPSC's commitments and found them acceptable. The discharge pressure switches were subsequently installed.

In January and February of 2005 while NMC was researching the basis for the AFW pump low discharge pressure switch setpoints, questions arose regarding the sufficiency of these switches to protect the AFW pumps in the case of a loss of suction caused by a seismic event or tornado.

The current function of the AFW pump low discharge pressure trip is to protect the AFW pump from damage caused by cavitation resulting from a loss of the non-safety related normal suction source, since portions of the normal suction supply are not qualified to the design basis earthquake or the design basis tornado. The safety related backup supply, the Service Water (SW) system, does not require the AFW pumps to have protection from a loss of suction because the SW system supply to the AFW pumps is seismic and tornado protected. In late February 2005, it was determined that the low discharge pressure trips would not perform the intended protection function. During subsequent review of the AFW system, it was also determined, using hydraulic models, that the AFW pumps were not adequately protected from a runout condition.

Analyses have been performed to verify the adequacy of the design and modifications to ensure the AFW pumps continue to perform their intended safety function. NMC requests NRC review and approval of the TS and licensing basis changes proposed in this LAR to support the associated plant modifications to AFW system design and operation.

4.0 TECHNICAL ANALYSIS

KNPP is a single unit plant located on the west bank of Lake Michigan approximately 10 miles from Kewaunee, Wisconsin. The facility is owned by WPSC and Wisconsin Power and Light Company and operated by the Nuclear Management Company (NMC). The unit at KNPP employs a two-loop pressurized water reactor designed and supplied by Westinghouse Electric Corporation. The initial KNPP application for a Construction Permit and Operating License was submitted to the Atomic Energy Commission (AEC) in August 1967. The Final Safety Analysis Report (FSAR) was submitted for application of an Operating License in January 1971. KNPP began commercial operation in December 1973.

The KNPP was designed and constructed to comply with WPSC's understanding of the intent of the AEC General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967. KNPP was not licensed to NUREG-0800, "Standard Review Plan (SRP)."

AFW System Design

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply.

The Auxiliary Feedwater System is shown in the KNPP Updated Safety Analysis Report (USAR) in Figure 10.2-3. The system consists of one steam turbine-driven pump and two motor-driven pumps, each capable of delivering feedwater to either or both steam generators. Separate safeguards buses provide electrical power to the two motor-driven pumps. Each of the three pumps is identical in design and capable of supplying approximately 200 gallons per minute (gpm) to the steam generator(s).

The normal water supply to the auxiliary feedwater pumps is from two 75,000-gallon condensate storage tanks (CSTs). The Class I Service Water (SW) System provides a backup safety related water supply to the pumps. An individual trip for each auxiliary feedwater pump is initiated on low pump discharge pressure. This trip was intended to provide automatic protection for the pumps in the event the condensate supply to the pumps is lost. Following a pump trip, the pumps can be restarted following manual switchover to the SW System.

The Auxiliary Feedwater System does not have functional requirements during normal plant operation. The system is used during hot standby conditions if the main feedwater pumps are not operating. Feedwater is required in this condition because the steam generator is used to remove residual heat or cool down the Reactor Coolant System until the Residual Heat Removal System (RHR) is capable of providing the necessary heat sink for the Reactor Coolant System residual heat.

During abnormal conditions, the Auxiliary Feedwater System provides three essential functions:

1. prevents thermal cycling of the steam-generator tube sheet upon loss of the main feedwater pump,

2. removes residual heat via the steam generators from the Reactor Coolant System until the RHR system is capable of providing the necessary heat sink, and

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

For the AFW system to meet these functions, the system must be able to be started, both manually and automatically, and provide flow to the steam generators.

The auxiliary feedwater pumps start on the following:

a. Motor-Driven Pumps

   - Safety Injection
   - Blackout
   - Low-Low level (2/3) in either steam generator starts both pumps
   - AMSAC low-low level (3/4) in both steam generators starts both pumps
   - Opening of both feedwater pump circuit breakers
   - Manual

b. Turbine-Driven Pump

- Low-low level in both steam generators
- AMSAC low-low level (3/4) in both steam generators
- Loss of voltage on both 4-KV buses (Bus 1 & Bus 2)
- Manual

With power less than or equal to 1673 MWth, each AFW pump provides 100% of the required capacity to the steam generators following a reactor trip, as assumed in the accident analysis. With power greater than 1673 MWth, two AFW pumps are required. Since the Auxiliary Feedwater System is an engineered safety features system, a backup pump is provided.

Current TS, Licensing Basis and Problem Definition

The current KNPP TS require that each auxiliary feedwater pump low discharge pressure trip be operable. If a trip channel is made or found to be inoperable, a completion time of 4 hours to return the channel to an operable status is allowed. If a channel is inoperable, for greater than 4 hours, the associated auxiliary feedwater train is declared inoperable and requirements of TS 3.4.b.3 and TS 3.4.b.4 are applied. TS 3.4.b.3 requires reactor power to be limited to 1673 MWth if two of the three AFW pumps are inoperable and TS 3.4.b.4 states the completion times for returning inoperable AFW pumps to service. If the operability requirements of TS 3.4.b.4 are not met, then TS 3.4.b.6 is entered. TS 3.4.b.6 states the required shutdown actions and associated completion times to place the plant in a condition where the reactor coolant temperature is less than 350°F.

During the January-February 2005 evaluations of AFW system design and performance capability, NMC determined that following a seismic, tornado or HELB event resulting in a rupture of the CST or CST supply piping to the AFW pumps, the AFW pump discharge pressure trip would not adequately protect the pump. A hydraulic review of the suction piping system showed that, if the suction piping fails, air could enter into the pump suction and potentially lead to pump damage. The damage could occur before the AFW pumps could be manually transferred to the qualified SW suction water source. Therefore, failure of the CSTs or the suction piping from the CSTs could result in damage to all three AFW pumps.

NMC also performed a review of the operating characteristics of the AFW pumps during design basis postulated accident scenarios. This review determined that the AFW pumps had a potential to be damaged during a depressurized or faulted steam generator event. The AFW pump(s) could be damaged by operating in a runout condition during steam generator depressurization.

During the Main Steam Line Break accident (USAR Section 14.2.5) low steam generator pressures may result in tripping of the AFW pumps. The MSLB analysis assumes maximum AFW flow as a conservative assumption providing greater mass for the steam release. Per the analysis, AFW flow is not required. Therefore, the current safety analysis bounds the tripping of the AFW pumps. The need for AFW is limited to supporting long-term decay heat removal and there is ample time to manually start and align the system to support this post accident function. These actions will be proceduralized and time validated.

During the Steam Generator Tube Rupture (SGTR) (USAR Section 14.2.4) radiological consequences are analyzed assuming a continuous release. All cases consider the same post-trip RCS pressure of 1930 psia and post-trip steam generator pressure of 895.6 psia. The highest post-trip flashing fraction, based on the range of operating temperatures covered by the analysis, is for a case with T-hot of approximately 606.8°F. It is conservatively assumed that the T-hot is not reduced for the 30 minutes in which the break flow is calculated. AFW flow is considered to be operating for 30 minutes. None of the assumptions or results in the SGTR analysis are affected by this modification.

Additionally, through the use of the normal remote air operated discharge control valves, the motor driven AFW pumps will remain capable of being operated consistent with the recovery procedure specified in Section 14.2.4 of the USAR. Should these air operated valves fail in the open position or the turbine driven AFW pump be required to provide AFW flow, local manual action will be taken to throttle the pump discharge isolation valves to maintain pump discharge pressure above pump trip setpoints. This would be an anticipated action in response to plant cooldown activities. These actions will be proceduralized and time validated. However, these actions are not addressed within the analysis in Section 14.2.4 of the USAR.

When reactor power is <15% of rated power, including operations involving potentially low steam generator pressures, the automatic start of AFW pumps is not an important to safety function (Technical Specification Bases 3.4.b). Therefore, an initial trip due to failure to correctly throttle discharge valves is easily recoverable and is therefore not risk significant. These actions will be proceduralized.

Proposed TS and Licensing Basis Changes and Problem Resolution

NMC's review of AFW system performance identified two issues which require resolution: 1) loss of suction protection; and 2) runout protection.

Suction Protection

NMC determined that the loss of suction protection issue will be resolved through the addition of a low suction pressure trip combined with an additional water volume in the AFW pump normal suction line. The additional suction water volume provides sufficient reserve water volume to allow the AFW pump to coast to a stop following the trip signal, thus preventing AFW pump damage. The AFW discharge pressure trip will no longer be used for loss of suction protection. For consistency with current TS, NMC proposes to include the new low suction pressure trip in the KNPP TS. The suction pressure trip instrumentation being added is similar in design to the existing discharge pressure trip instrumentation.

Suction pressure switch failure mechanisms considered included a) pressure switch failure, b) instrument tubing failure, and c) inadvertent pressure switch actuation. Time delay relay failure mechanisms considered included a) time delay relay failure, and b) inadvertent actuation of the time delay relay. Suction pressure trip bypass switch failure mechanisms considered included a) bypass switch failure, and b) inadvertent bypass switch actuation. The new instrumentation, installed as designed and according to approved procedures, satisfies KNPP design, quality and separation requirements. The evaluation of the new suction pressure trip circuit design concluded that the new suction pressure trip circuit is similar to the existing discharge pressure trip circuit design and therefore, no new failure modes or effects are introduced.

For a loss of normal suction, an additional water volume will be established in a new section of suction piping that is qualified to withstand a seismic event or a tornado as required for Nuclear Safety Design Class I (Class I) systems, structures, or components. The design of the Class I AFW pump suction piping is such that if the portion of the suction piping that is non-Class I were to fail (i.e., break due to a seismic event or tornado) the pumps would be protected. Plant analyses have demonstrated that the suction pressure trip combined with an additional suction water volume protect the AFW pumps by sensing the loss of suction, tripping the associated AFW pump, and providing sufficient reserve water volume to allow the AFW pump to coast to a stop thus preventing AFW pump damage.

Runout Protection

NMC determined that AFW pump runout protection will be resolved through use of low discharge pressure trips with setpoints that ensure adequate protection for pump runout conditions. The low discharge pressure switches on the discharge of the AFW pumps will have a new function to protect the AFW pumps from damage due to inadequate net positive suction head (NPSH) in pump runout conditions. The discharge pressure switches will no longer be credited for protection of the pumps under a loss of CST supply.

Because the discharge pressure switch setpoint is increased to protect an AFW pump from runout conditions and the flow control valves have non-safety related controls, additional plant operator actions may be required. Specifically, if necessary, an operator will be dispatched to the AFW pump room(s) to locally throttle manual valves in the discharge of the AFW pumps. This action would establish adequate AFW pump discharge backpressure on pump restart to prevent runout, while allowing the local operator, in communication with the Control Room, to regulate flow to the desired flow rate. Local discharge pressure indication is also available to the local operator, providing immediate monitoring capability.

A review was performed to determine the impact on the accident analyses with an increase in the AFW low discharge pressure trip setpoint from 350 psig for the motor-driven AFW Pumps and 100 psig for the turbine-driven AFW pumps, to 860 psig for all the AFW pumps' low discharge pressure trip setpoint. Engineering analysis shows that if the steam generator's pressure remains above 604 psig the AFW pumps will not trip due to the pumps discharge pressure being greater than the low discharge pressure trip setpoint. A review showed that for the time period of analysis for the USAR transients and accidents, SG pressures are greater than 750 psig for the duration of each event, except the main steam line break (MSLB) inside containment and the HELB-MSLB outside containment events. However, during the steam generator tube rupture event analyzed in Chapter 14, operator actions to cool the RCS by depressurizing the steam generators may cause the AFW pumps to trip due to low discharge pressure.

During the above events or recovery actions where SG pressure drops below 604 psig, the low discharge pressure switches could trip the AFW pumps. NMC has evaluated the requirement for operator actions to restore the AFW pumps for post-accident recovery activities. Plant operator actions, from the Control Room or locally, are necessary to realign, re-initiate and control AFW flow following AFW pump trips as a result of low discharge pressure (runout protection) subsequent to a steam generator depressurization scenario such as MSLB or SGTR recovery actions.

Existing procedures direct plant operator actions, from the Control Room, for those scenarios where the AFW pumps trip and must be restarted. Faulted SG isolation is directed through closure of safety related cross-connect motor-operated valve(s). Subsequently, and prior to AFW pump restart, AFW discharge flow is throttled through the use of safety related AFW pump discharge flow control valves with non-safety related controls.

During post-accident recovery operations, the plant emergency procedures will re-establish AFW system availability. These operator actions to reconfigure the AFW system are recovery actions. Engineering review performed as a part of this modification determined that the AFW pump manual discharge and cross connect valves can be repositioned, as required. Post-modification testing will include throttling the motor driven AFW pump discharge gate valves (AFW-3A and AFW-3B) to verify the ability to manually throttle the valves with high differential pressure across the valve disk. Throttling the motor driven AFW pump valves will bound the TD AFW pump discharge valve as the differential pressure across AFW-3A and AFW-3B will be higher that that seen across the TDAFW pump discharge gate valve (AFW-2C). This is because the TDAFW pump requires approximately 300 psi in the steam generator to develop full speed, whereas the motor driven AFW pumps will always develop full speed and discharge pressure and therefore, as these three valves are the same model valve (i.e., 3 inch diameter William Powell, 19000 series gate valves) the motor driven conditions are bounding. In conjunction with procedure revisions, operator actions will be time validated to ensure the AFW system performs consistent with the licensing and design basis. The modification process tracks completion of these activities. The AFW system is necessary during the post-accident recovery operations to control temperature and potential pressurization of the RCS.

Conclusions

NMC has performed analyses to demonstrate that the proposed TS and licensing basis changes in conjunction with the plant modifications will assure that the AFW system remains available to perform its required design safety function. Operation of the Kewaunee Nuclear Power Plant with these licensing basis changes and revised TS will continue to protect the health and safety of the public.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

Nuclear Management Company, LLC (NMC) proposes Technical Specification (TS) changes to modify the auxiliary feedwater (AFW) pump suction protection requirements. The proposed TS changes include 3.4.b, "Auxiliary Feedwater System" and TS Table 4.1-1, "Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels." NMC also proposes licensing basis changes to revise the functionality of the discharge pressure switches to provide pump runout protection, which requires operator actions to restore the AFW pumps for specific post-accident recovery activities.

NMC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed amendment does not involve a significant increase in the probability of an accident previously evaluated. The proposed changes are associated with the auxiliary feedwater (AFW) system, which is not an initiator of any accident previously evaluated.

   The proposed amendment does not involve a significant increase in the consequences of an accident previously evaluated. The mitigation functions assumed in the accident analyses will continue to be performed. Operator actions may be required to assure the AFW pumps are aligned for post-accident recovery operations. With these actions additional consequences are not incurred.

Therefore, operation of the facility in accordance the proposed amendment would not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The AFW system is being modified by adding suction pressure switches to protect the AFW pumps from damage due to a loss of normal suction. The addition of the suction pressure switches and the associated circuitry does not introduce new failure modes or effects. The evaluation of the new suction pressure trip circuit design concluded the new suction pressure trip circuit is similar to the existing discharge pressure trip circuit design and therefore, no new failure modes or effects are introduced. In addition, the AFW system is being modified by altering the function of the discharge pressure trip channel to provide pump runout protection. Operator actions may be required to assure the AFW pumps are aligned for post-accident recovery operations. With these actions, the accident recovery operations can be performed and a new or different kind of accident is not created. The proposed amendment ensures that the AFW system continues to performs its intended safety function.

   Therefore, operation of the facility in accordance with the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The modifications to the AFW System and the associated Technical Specifications will ensure that the AFW system is capable of performing its intended safety function. In addition, the margin of safety in the accident analyses is not affected by the proposed changes. The manual actions that may be required to restart an AFW pump and throttle AFW flow during the cooldown/recovery phase of the event do not significantly impact the margin of safety.

   Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the evaluation above, NMC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

General Design Criteria Evaluation

The US Atomic Energy Commission (AEC) issued their Safety Evaluation (SE) of the Kewaunee Nuclear Power Plant on July 24, 1972, with supplements dated December 18, 1972 and May 10, 1973. The AEC's SE, section 3.1, "Conformance with AEC General Design Criteria," described the conclusions the AEC reached associated with the General Design Criteria in effect at the time. The AEC stated:

> *The Kewaunee plant was designed and constructed to meet the intent of the AEC's General Design Criteria, as originally proposed in July 1967. Construction of the plant was about 50% complete and the Final Safety Analysis Report (Amendment No. 7) had been filed with the Commission before publication of the revised General Design Criteria in February 1971 and the present version of the criteria in July 1971. As a result, we did not require the applicant to reanalyze the plant or resubmit the FSAR. However, our technical review did assess the plant against the General Design Criteria now in effect and we are satisfied that the plant design generally conforms to the intent of these criteria.*

As such the appropriate 10 CFR 50 Appendix A General Design Criteria are listed below with the associated criteria KNPP is licensed to from the Final Safety Analysis (Amendment 7), which has been updated and now titled the Updated Safety Analysis Report (USAR). Below are the applicable KNPP General Design Criterion (GDC).

The AFW system is considered an Engineered Safety Feature (ESF) system at the KNPP. As such, the following GDC apply.

GDC 37 - Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. Such engineered safety features shall be designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary, assuming unobstructed discharge from both ends.

Answer: The Containment System, the Containment Isolation System, the Emergency Core Cooling System, the Special Zone Ventilation Systems, the Containment Vessel Internal Spray System, the Auxiliary Feedwater System, the diesel generators, and the station batteries comprise the Engineered Safety Features for the facility. These systems and their supporting systems (Component Cooling System and Service Water System) are designed to cope with any size reactor coolant pressure boundary break, up to and including rupture of the largest reactor coolant pipe.

GDC 40 - Adequate protection for those engineered safety features, the failure of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures

Answer: All engineered safety features are protected against dynamic effects and missiles resulting from equipment failures.

Based on the review of the applicable general design criteria, the AFW system, as modified, continues to meet these general design criteria.

NUREG-0800, Standard Review Plan

Although KNPP is not committed to NUREG-0800, the plant-specific AFW system designs were evaluated against NUREG-0800, "Standard Review Plan," for applicable guidance. Specifically, the AFW system design was evaluated against the following NUREG-0800 sections: "Standard Review Plan," Section 10.4.9, "Auxiliary Feedwater System," including Branch Technical Position Auxiliary Systems Branch 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants." NMC concluded that the AFW system, with the recent modifications, proposed TS changes and proposed licensing basis changes meets the intent of NUREG-0800 guidance.

Improved Standard Technical Specification Comparison

NMC has not converted the KNPP TS to conform to the format and guidance of NUREG-1431, Standard Technical Specifications, Westinghouse Plants, (ISTS). However, NUREG-1431 was consulted for applicable guidance. The ISTS 3.3.2 instrumentation LCOs apply to inputs to the AFW System, but the instrumentation of interest in this LAR is protective instrumentation for the AFW pumps which is internal to the system. Thus the guidance in ISTS LCO 3.3.2 is not applicable.

Regulatory Requirements/Criteria Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

ENCLOSURE 2
NUCLEAR MANAGEMENT COMPANY, LLC ,
MARKED UP TS PAGES FOR LICENSE AMENDMENT REQUEST 213 TO
KEWAUNEE NUCLEAR POWER PLANT, OPERATING LICENSE NO. DPR-43,
DOCKET NO. 50-305

Marked Up TS Pages:

TS 3.4-1
TS 3.4-2
Table TS 4.1-1, Page 7 of 7

3 Pages to Follow

3.4 STEAM AND POWER CONVERSION SYSTEM

APPLICABILITY

Applies to the OPERATING status of the Steam and Power Conversion System.

OBJECTIVE

To assure minimum conditions of steam-relieving capacity and auxiliary feedwater supply necessary to assure the capability of removing decay heat from the reactor, and to limit the concentrations of water activity that might be released by steam relief to the atmosphere.

SPECIFICATION

a. Main Steam Safety Valves (MSSVs)

1. The Reactor Coolant System shall not be heated > 350°F unless a minimum of two MSSVs per steam generator are OPERABLE.

2. The reactor shall not be made critical unless five MSSVs per steam generator are OPERABLE.

3. If the conditions of TS 3.4.a.1 or TS 3.4.a.2 cannot be met within 48 hours, then 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.

b. Auxiliary Feedwater System

1. The Reactor Coolant System shall not be heated > 350°F unless the following conditions are met:

   A. Auxiliary feedwater train "A" and auxiliary feedwater train "B" are OPERABLE and capable of taking suction from the Service Water System and delivering flow to the associated steam generator.

   B. The turbine-driven auxiliary feedwater train is OPERABLE and capable of taking suction from the Service Water System and delivering flow to both steam generators, OR

   The turbine-driven auxiliary feedwater train is declared inoperable.

   C. The auxiliary feedwater pump low discharge pressure trip channels are OPERABLE.

   D. The auxiliary feedwater pump low suction pressure trip channels are OPERABLE.

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 ≤ 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~~ one auxiliary feedwater pump's low discharge pressure trip channel and/or ~~a~~ low suction pressure trip channel may be inoperable for a period not to exceed 4 hours. If this time period is exceeded or more than one pump's trip channel(s) are inoperable then the associated auxiliary feedwater train(s) 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.

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
43. AFW Pump Low Discharge Pressure Trip	Not Applicable	Each refueling cycle	Quarterly~~Each refueling cycle~~ (a)	(a) Verification of relay setpoints not required.
44. Axial Flux Difference (AFD)	Weekly			Verify AFD within limits for each OPERABLE excore channel
45. Service Water Turbine Header Isolation Logic Trip (SW 4 A/B)	Not Applicable	Each refueling cycle	Each refueling cycle	
46. AFW Pump Low Suction Pressure Trip	Not Applicable	Each refueling cycle	Quarterly (a)	(a) Verification of relay setpoints not required.

ENCLOSURE 3
NUCLEAR MANAGEMENT COMPANY, LLC,
AFFECTED TS PAGES FOR LICENSE AMENDMENT REQUEST 213 TO
KEWAUNEE NUCLEAR POWER PLANT, OPERATING LICENSE NO. DPR-43,
DOCKET NO. 50-305

Affected TS Pages:

TS 3.4-1
TS 3.4-2
Table TS 4.1-1, Page 7 of 7

3 Pages to Follow

3.4 STEAM AND POWER CONVERSION SYSTEM

APPLICABILITY

Applies to the OPERATING status of the Steam and Power Conversion System.

OBJECTIVE

To assure minimum conditions of steam-relieving capacity and auxiliary feedwater supply necessary to assure the capability of removing decay heat from the reactor, and to limit the concentrations of water activity that might be released by steam relief to the atmosphere.

SPECIFICATION

a. Main Steam Safety Valves (MSSVs)

1. The Reactor Coolant System shall not be heated > 350°F unless a minimum of two MSSVs per steam generator are OPERABLE.

2. The reactor shall not be made critical unless five MSSVs per steam generator are OPERABLE.

3. If the conditions of TS 3.4.a.1 or TS 3.4.a.2 cannot be met within 48 hours, then 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.

b. Auxiliary Feedwater System

1. The Reactor Coolant System shall not be heated > 350°F unless the following conditions are met:

   A. Auxiliary feedwater train "A" and auxiliary feedwater train "B" are OPERABLE and capable of taking suction from the Service Water System and delivering flow to the associated steam generator.

   B. The turbine-driven auxiliary feedwater train is OPERABLE and capable of taking suction from the Service Water System and delivering flow to both steam generators, OR

      The turbine-driven auxiliary feedwater train is declared inoperable.

   C. The auxiliary feedwater pump low discharge pressure trip channels are OPERABLE.

   D. The auxiliary feedwater pump low suction pressure trip channels are OPERABLE.

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 ≤ 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, one auxiliary feedwater pump's low discharge pressure trip channel and/or low suction pressure trip channel may be inoperable for a period not to exceed 4 hours. If this time period is exceeded or more than one pump's trip channel(s) are inoperable then the associated auxiliary feedwater train(s) 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.

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
43. AFW Pump Low Discharge Pressure Trip	Not Applicable	Each refueling cycle	Quarterly (a)	(a) Verification of relay setpoints not required.
44. Axial Flux Difference (AFD)	Weekly			Verify AFD within limits for each OPERABLE excore channel
45. Service Water Turbine Header Isolation Logic Trip (SW 4 A/B)	Not Applicable	Each refueling cycle	Each refueling cycle	
46. AFW Pump Low Suction Pressure Trip	Not Applicable	Each refueling cycle	Quarterly (a)	(a) Verification of relay setpoints not required.

ENCLOSURE 4
NUCLEAR MANAGEMENT COMPANY, LLC,
MARKED UP TS BASIS PAGES FOR LICENSE AMENDMENT REQUEST 213 TO KEWAUNEE NUCLEAR POWER PLANT, OPERATING LICENSE NO. DPR-43, DOCKET NO. 50-305

Marked Up TS Basis Pages

TS B3.4-1
TS B3.4-2
TS B3.4-3
TS B3.4-4
TS B3.4-5

5 Pages to Follow

BASIS - Steam and Power Conversion System (TS 3.4)

Main Steam Safety Valves (TS 3.4.a)

The ten main steam safety valves (MSSVs) (five per steam generator) have a total combined rated capability of 7,660,380 lbs./hr. at 1181 lbs./in.^{2} pressure. This flow ensures that the main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by ASME B&PV Code) for the worst-case loss-of-sink-event.

While the plant is in the HOT SHUTDOWN condition, at least two main steam safety valves per steam generator are required to be available to provide sufficient relief capacity to protect the system.

The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Plan.

Auxiliary Feedwater System (TS 3.4.b)

The Auxiliary Feedwater (AFW) System is designed to remove decay heat during plant startups, plant shutdowns, and under accident conditions. During plant startups and shutdowns the system is used in the transition between Residual Heat Removal (RHR) System decay heat removal and Main Feedwater System operation.

The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow from the AFW pumps to the steam generators are OPERABLE. This requires that the two motor-driven AFW pumps be OPERABLE, each capable of taking suction from the Service Water System, capable of discharge throttling with AFW-3A or AFW-3B, and supplying AFW to separate steam generators (SGs). The turbine-driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the main steam isolation valves and shall be capable of taking suction from the Service Water System, capable of discharge throttling with AFW-2C, and supplying AFW to both of the steam generators. With no AFW trains OPERABLE, immediate action shall be taken to restore a train.

Auxiliary feedwater trains are defined as follows:

"A" train -	"A" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "A" steam generator, not including AFW-10A or AFW-10B
"B" train -	"B" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "B" steam generator, not including AFW-10A or AFW-10B
Turbine-driven train -	Turbine-driven AFW pump and associated AFW valves and piping to both "A" steam generator and "B" steam generator, including AFW-10A and AFW-10B

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 <u>via the steam generators</u> 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.

<u>TS 3.4.b.5</u>

<u>AFW Pump Low Discharge Pressure Trip</u>

<u>This Function must be OPERABLE when the average RCS temperature is > 350°F to ensure that the AFW System is available to maintain the SGs as the heat sink for the reactor. This Function does not have to be OPERABLE when the average RCS temperature is ≤ 350°F because RHR is required to be OPERABLE to remove decay heat.</u>

<u>A low discharge pressure signal in the AFW pump discharge line protects the AFW pumps from damage due to runout conditions during alignment and operation of the pumps to a depressurized steam generator. A low-pressure signal sensed by any one of the pump switches will cause the associated AFW pump to trip. Operator action is required to align the associated pump to the non-faulted steam generator, throttle the AFW pump discharge flow, if necessary, and restart the pump(s).</u>

<u>AFW Pump Low Suction Pressure Trip</u>

<u>This Function must be OPERABLE when the average RCS temperature is > 350°F to ensure that the AFW System is available for alignment to Service Water for the AFW System to maintain the SGs as the heat sink for the reactor. This Function does not have to be OPERABLE when the average RCS temperature is ≤ 350°F because RHR is required to be OPERABLE to remove decay heat.</u>

<u>A low pressure signal in the AFW pump suction line protects the AFW pumps against a loss of the normal water supply from the condensate storage tanks (CSTs). Three pressure switches are located on the AFW pump suction line from the CST. A low-pressure signal sensed by any one of the switches will cause the associated AFW pump to trip. Operator action is required to bypass the trip circuit or align to the Service Water source and restart the associated AFW pump. Service Water alignment and restart of the AFW pumps ensures an adequate supply of water to maintain at least one of the SGs as the heat sink for reactor decay heat and sensible heat removal.</u>

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

ENCLOSURE 5
NUCLEAR MANAGEMENT COMPANY, LLC,
AFFECTED TS BASIS PAGES FOR LICENSE AMENDMENT REQUEST 213 TO KEWAUNEE NUCLEAR POWER PLANT OPERATING LICENSE NO. DPR-43, DOCKET NO. 50-305

Affected TS Basis Pages

TS B3.4-1
TS B3.4-2
TS B3.4-3
TS B3.4-4
TS B3.4-5

5 Pages to Follow

BASIS - Steam and Power Conversion System (TS 3.4)

Main Steam Safety Valves (TS 3.4.a)

The ten main steam safety valves (MSSVs) (five per steam generator) have a total combined rated capability of 7,660,380 lbs./hr. at 1181 lbs./in.^{2} pressure. This flow ensures that the main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by ASME B&PV Code) for the worst-case loss-of-sink-event.

While the plant is in the HOT SHUTDOWN condition, at least two main steam safety valves per steam generator are required to be available to provide sufficient relief capacity to protect the system.

The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Plan.

Auxiliary Feedwater System (TS 3.4.b)

The Auxiliary Feedwater (AFW) System is designed to remove decay heat during plant startups, plant shutdowns, and under accident conditions. During plant startups and shutdowns the system is used in the transition between Residual Heat Removal (RHR) System decay heat removal and Main Feedwater System operation.

The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow from the AFW pumps to the steam generators are OPERABLE. This requires that the two motor-driven AFW pumps be OPERABLE, each capable of taking suction from the Service Water System, capable of discharge throttling with AFW-3A or AFW-3B, and supplying AFW to separate steam generators (SGs). The turbine-driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the main steam isolation valves and shall be capable of taking suction from the Service Water System, capable of discharge throttling with AFW-2C, and supplying AFW to both of the steam generators. With no AFW trains OPERABLE, immediate action shall be taken to restore a train.

Auxiliary feedwater trains are defined as follows:

"A" train -	"A" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "A" steam generator, not including AFW-10A or AFW-10B
"B" train -	"B" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "B" steam generator, not including AFW-10A or AFW-10B
Turbine-driven train -	Turbine-driven AFW pump and associated AFW valves and piping to both "A" steam generator and "B" steam generator, including AFW-10A and AFW-10B

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 via the steam generators 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.

<u>TS 3.4.b.5</u>

<u>AFW Pump Low Discharge Pressure Trip</u>

This Function must be OPERABLE when the average RCS temperature is > 350°F to ensure that the AFW System is available to maintain the SGs as the heat sink for the reactor. This Function does not have to be OPERABLE when the average RCS temperature is ≤ 350°F because RHR is required to be OPERABLE to remove decay heat.

A low discharge pressure signal in the AFW pump discharge line protects the AFW pumps from damage due to runout conditions during alignment and operation of the pumps to a depressurized steam generator. A low-pressure signal sensed by any one of the pump switches will cause the associated AFW pump to trip. Operator action is required to align the associated pump to the non-faulted steam generator, throttle the AFW pump discharge flow, if necessary, and restart the pump(s).

<u>AFW Pump Low Suction Pressure Trip</u>

This Function must be OPERABLE when the average RCS temperature is > 350°F to ensure that the AFW System is available for alignment to Service Water for the AFW System to maintain the SGs as the heat sink for the reactor. This Function does not have to be OPERABLE when the average RCS temperature is ≤ 350°F because RHR is required to be OPERABLE to remove decay heat.

A low pressure signal in the AFW pump suction line protects the AFW pumps against a loss of the normal water supply from the condensate storage tanks (CSTs). Three pressure switches are located on the AFW pump suction line from the CST. A low-pressure signal sensed by any one of the switches will cause the associated AFW pump to trip. Operator action is required to bypass the trip circuit or align to the Service Water source and restart the associated AFW pump. Service Water alignment and restart of the AFW pumps ensures an adequate supply of water to maintain at least one of the SGs as the heat sink for reactor decay heat and sensible heat removal.

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