June 22, 1990

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Docket No. 50-305

Mr. Ken H. Evers
Manager - Nuclear Power
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, Wisconsin 54307-9002

Dear Mr. Evers:

SUBJECT: AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. DPR-43 (TAC NO. 76036)

The Commission has issued the enclosed Amendment No. 87 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications in response to your application dated February 19, 1990.

The amendment deletes TS 4.3 concerning reactor coolant system (RES) leak testing and weld examination requirements. These requirements are covered in TS 4.2a, which requires leak testing and inservice inspection be conducted in accordance with Section XI of the KSME Boiler and Pressure Vessel Code.

The amendment also revises TS 3.3.d.2 and 3.3.e.2 to eliminate the requirement to place the plant in a cold shutdown condition upon a long-term loss of one train of component cooling water (CCW) or service water (SW). This will prevent having to place the plant in a mode which would demand the maximum support from the CCW or SW systems. By maintaining RCS temperature above 200°F but less than 350°F, the steam generators would be available as an additional means of decay heat removal.

Also included in this amendment are several miscellaneous revisions that correct typographical errors, correct inconsistencies, and clarify the intent of certain technical specifications.

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A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

/S/

Michael J. Davis, Project Manager
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 87 to License No. DPR-43
2. Safety Evaluation

cc w/enclosures:
See next page

| | | | | | |
|---|---|---|---|---|---|
| Office: | LA/PDIII-3 | PM/PDIII-3 | PD/PDIII-3 | OGC-WF1 | EMTB |
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| Date: | 4/27/90 4/26 | 4/27/90 | 6/13/90 | 6/18/90 | 6/4/90 |

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 22, 1990

Docket No. 50-305

Mr. Ken H. Evers
Manager - Nuclear Power
Wisconsin Public Service Corp.
P.O. Box 19002
Green Bay, Wisconsin 54037-9002

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Also included in this amendment are several miscellaneous revisions that correct typographical errors, correct inconsistencies, and clarify the intent of certain technical specifications.

A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

for Thomas V. Wambach

Michael J. Davis, Project Manager
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 87 to
   License No. DPR-43
2. Safety Evaluation

cc w/enclosures:
See next page

cc:
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Town of Carlton
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Hill Farms State Office Building
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
License No. DPR-43

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Wisconsin Public Service Corporation, Wisconsin Power and Light Company, and Madison Gas and Electric Company (the licensees) dated February 19, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-43 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 87, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, and is to be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John N. Hannon

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 22, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 87

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

| REMOVE | INSERT |
|---|---|
| TS iii | TS iii |
| TS 3.2-1 | TS 3.2-1 |
| TS 3.2-2 | TS 3.2-2 |
| TS 3.2-3 | TS 3.2-3 |
| TS 3.3-6 | TS 3.3-6 |
| TS 3.3-7 | TS 3.3-7 |
| TS 3.3-9 | TS 3.3-9 |
| TS 3.5-3 | TS 3.5-3 |
| TS 3.5-4 | TS 3.5-4 |
| TS 4.2-2 | TS 4.2-2 |
| TS 4.2-9 | TS 4.2-9 |
| TS 4.3-1 | TS 4.3-1 |
| TS 4.3-2 | TS 4.3-2 |
| TS 4.4-7 | TS 4.4-7 |
| TS 4.4-9 | TS 4.4-9 |
| TS 4.5-2 | TS 4.5-2 |

 Amendment No. ~~64~~,~~76~~,~~81~~, 87,

## 3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability

Applies to the operational status of the Chemical and Volume Control System.

Objective

To define those conditions of the Chemical and Volume Control System necessary to ensure safe reactor operation.

Specification

a. When fuel is in the reactor there shall be at least one flow path to the core for boric acid injection.

b. The reactor shall not be made critical unless the following conditions are satisfied, except as provided in Specification 3.2.c.

1. A minimum of TWO charging pumps shall be operable.

2. BOTH boric acid transfer pumps shall be operable.

3. At least ONE boric acid tank shall contain a minimum of 2,000 gallons of 11.5% to 13% by weight boric acid (19,700 to 23,000 ppm boron) solution at a temperature of at least 145°F.

4. System piping and valves shall be operable to the extent of establishing flow paths from the boric acid tank(s) and the Refueling Water Storage Tank to the Reactor Coolant System.

5. TWO trains of heat tracing shall be operable for the above flow paths for concentrated boric acid.

c. Any one of the following conditions of inoperability may exist during the time intervals specified. The reactor shall be placed in the hot shutdown condition if operability is not restored within the time specified, and it shall be placed in the cold shutdown condition if operability is not restored within an additional 48 hours.

   1. ONE of the operable charging pumps may be removed from service provided two pumps are again operable within 24 hours.

   2. ONE boric acid transfer pump may be out of service provided both pumps are operable within 24 hours.

   3. ONE train of heat tracing may be out of service provided it is restored to operable status within 48 hours.

<u>Basis</u>

The Chemical and Volume Control System provides control of the Reactor Coolant System boron inventory. This is normally accomplished by using any one of the three charging pumps in series with any one of the two boric acid transfer pumps. An alternate method of boration will be use of the charging pumps directly from the Refueling Water Storage Tank. A third method will be to use the safety injection pumps. There are two sources of borated water available for injection through 3 different paths.

(1) The boric acid transfer pumps can deliver the boric acid tank contents to the suction of the charging pumps.

(2) The charging pumps can take suction directly from the Refueling Water Storage Tank containing a concentration of 1950 ppm boron solution. Reference is made to Specification 3.3.b.1.A.

(3) The safety injection pumps can take their suctions from either the boric acid tanks or the Refueling Water Storage Tank.

The quantity of boric acid stored in either the boric acid tanks or the Refueling Water Storage Tank is sufficient to achieve cold shutdown at any time during core life.

Approximately 1800 gallons of boric acid of at least 11.5% concentration (19,700 ppm boron) is required to ensure cold shutdown. A minimum of 2000 gallons in the boric acid tank is therefore specified. A minimum temperature of 145°F is required to ensure solution solubility. Two trains of heat tracing are installed on lines normally containing concentrated boric acid solution.

The capacity of each charging pump is 60 gpm. This is sufficient to provide make-up water requirements for the reactor coolant system in the event of an allowable leak which permits continued safe plant operation. Any two of the three installed charging pumps can be used to comply with TS 3.2.b.1.

There are two trains of Boric Acid Heat Tracing with each train powered from a separate safeguard power supply, and each train being made up of several individual circuits. An individual circuit can be removed from service indefinitely, provided that the temperature of the fluid in that circuit can be maintained greater than 145°F without reliance on the redundant heat trace circuit.

d. Component Cooling System

1. The reactor shall not be made or maintained critical unless the following conditions are satisfied, except for low power physics tests and except as provided by Specification 3.3.d.2.

   A. TWO component cooling water trains are operable with each train consisting of:

      1. ONE component cooling water pump
      2. ONE component cooling water heat exchanger
      3. An operable flow path consisting of all valves and piping associated with the above train of components and required to function during accident conditions.

2. During power operation or recovery from an inadvertent trip, ONE component cooling water train may be inoperable for a period of 72 hours. If operability is not restored within 72 hours, 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 and maintain the Reactor Coolant System Tavg less than 350°F by use of alternate heat removal methods within an additional 36 hours

e. Service Water System

1. The reactor shall not be made critical unless the following conditions are satisfied, except for low-power physics tests and as provided by Specification 3.3.e.2.

   A. TWO service water trains are operable with each train cons of:

1. TWO service water pumps
2. 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 a suction from the forebay and supplying water to the redundant safeguards headers.

B. The forebay water level trip system is operable.

2. During power operation or recovery from an inadvertent trip, ONE service water train may be inoperable for a period of 72 hours. If operability is not restored within 72 hours, 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 and maintain Reactor Coolant System Tavg less than 350°F by use of alternate heat removal methods within an additional 36 hours.

Basis

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant.(1) With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore, to be conservative, most engineered safety features components and auxiliary cooling systems shall be fully operable.

requirements after a postulated loss-of-coolant accident. the malfunction(s) are not corrected after the specified time in a hot shutdown condition, the reactor will be placed in the cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would release fission products or damage the fuel elements.

When the inoperable component is part of the residual heat removal (RHR), component cooling water (CCW) or service water (SW) systems, the average reactor coolant system temperature ($T_{avg}$) will be maintained below 350°F through an alternate heat removal method. The various alternate heat removal methods include the redundant RHR train and the steam generators.

Assuming the reactor has been operating at full rated power for at least 100 days, the magnitude of the decay heat decreases as follows after initiating hot shutdown.

| Time After Shutdown | Decay Heat, % of Rated Power |
|---|---|
| 1 min. | 4.5 |
| 30 min. | 2.0 |
| 1 hour | 1.62 |
| 8 hours | 0.96 |
| 48 hours | 0.62 |

Thus the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safety features in order to effect repairs. Failure to complete repairs after placing the reactor in the hot shutdown con-

Containment Spray

Containment sprays are also actuated by a high containment pressure signal (Hi-Hi) to reduce containment pressure in the event of a loss-of-coolant or steam line break accident inside the containment.

The containment sprays are actuated at a higher containment pressure (approximately 50% of design containment pressure) than is safety injection (10% of design). Since spurious actuation of containment spray is to be avoided, it is initiated only on coincidence of high containment pressure (Hi-Hi) sensed by three sets of one-out-of-two containment pressure signals provided for its actuation.

Containment Isolation

A containment isolation signal is initiated by any signal causing automatic initiation of safety injection or may be initiated manually. The containment isolation system provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the outside environment in the event of a loss-of-coolant accident.

Steam Line Isolation

In the event of a steam line break, the steam line isolation valve of the affected line is automatically isolated to prevent continuous, uncontrolled steam release from more than one steam generator. The steam lines are isolated on Hi-Hi containment pressure or high steam flow in coincidence with Lo-Lo $T_{avg}$ and safety injection or Hi-Hi steam flow in coincidence with safety injection. Adequate protection is afforded for breaks inside or outside the containment even under the assumption that the steam line check valves do not function properly.

Setting Limits

1. The high containment pressure limit is set at about 10% of the maximum internal pressure. Initiation of Safety Injection protects against loss-of-coolant[2] or steam line break[3] accidents as discussed in the safety analysis.

2. The Hi-Hi containment pressure limit is set at about 50% of the maximum internal containment pressure for initiation of containment spray and at about 30% for initiation of steam line isolation. Initiation of containment spray and steam line isolation protects against large loss-of-coolant or steam line break accidents as discussed in the safety analysis.

3. The pressurizer low-pressure limit is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss-of-coolant accident as shown in the safety analysis.

4. The steam line low-pressure signal is lead/lag compensated and its set-point is set well above the pressure expected in the event of a large steam line break accident as shown in the safety analysis.

5. The high steam line flow limit is set at approximately 20% of nominal full-load flow at the no-load pressure and the high-high steam line flow limit is set at approximately 120% of nominal full load flow at the full load pressure in order to protect against large steam line break accidents. The coincident Lo-Lo $T_{avg}$ setting limit for steam line isolation initiation is set below its hot shutdown value. The safety analysis shows that these settings provide protection in the event of a large steam line break.

cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair, or replacement work is performed.

b. Whenever integrity of a pressure isolation valve listed in Table TS 3.1-2 cannot be demonstrated, the integrity of the remaining pressure isolation valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily.

(1) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

Basis

Technical Specification 4.2.b.2

Periodic inspection of the steam generator tubes allows evaluation of their service condition. As operational experience has become available it is evident that certain types of steam generators are susceptible to generic degradation mechanisms. Site specific steam generator tube degradation has also occurred throughout the industry. The inspection program at Kewaunee is designed to identify both generic and site specific tube degradation mechanisms.

Steam generator tube surveillance at Kewaunee is generally performed using eddy current techniques. Various methods of eddy current (EC) testing are used to inspect steam generator tubes for wall degradation. EC methods have improved considerably since Kewaunee began commercial operation in 1974. Single frequency EC testing with a single probe and X-Y plotter have evolved into multifrequency techniques with assorted probe types and sophisticated software to allow more accurate volumetric tube examinations. Profilometery techniques are also being developed which detect imperfections in a tube's original geometry. WPSC is committed to utilize advancing EC testing technology, as appropriate, to assure accurate determination of the steam generator tubes' service condition.

Technical Specification 4.2.b.3

Steam generator tube inspections are generally scheduled during refueling outages at the Kewaunee Nuclear Power Plant. The tubes scheduled for a given inspection are based upon their service condition determined during previous inspections, and operational experience from other plants with similar steam generators and water chemistry. Identification of degraded steam generator tube conditions results in augmentation of the inspection effort as well as increasing the frequency of subsequent inspections. In this manner, steam generator tube surveillance is consistent with service conditions.

There are several operational occurrences or transients that will require subsequent steam generator tube inspections. These inspections are required as a result of excessive primary to secondary leakage or transients imposing large mechanical and thermal stresses on the tubes.

-Deleted-

-Deleted-

d. Each train shall be operated with the heaters on at least 10 hours every month.

3. An air distribution test on these HEPA filter banks will be performed after any maintenance or testing that could affect the air distribution within the systems. The test shall be performed at design flow rate (±10%). The results of the test shall show the air distribution is uniform within ±20%.*

4. Each train shall be determined to be operable at the time of its periodic test if it produces measurable indicated vacuum in the annulus within two minutes after initiation of a simulated safety injection signal and obtains equilibrium discharge conditions that demonstrate the Shield Building leakage is within acceptable limits.

d. Auxiliary Building Special Ventilation System

1. Periodic tests of the Auxiliary Building Special Ventilation System, including the door interlocks, shall be performed in accordance with Specifications 4.4.c.1 through 4.4.c.3 except for Specification 4.4.c.2.d.
2. Each train of Auxiliary Building Special Ventilation System shall be operated with the heaters on at least 15 minutes every month.
3. Each system shall be determined to be operable at the time of periodic test if it starts with coincident isolation of the normal ventilation ducts and produces a measurable vacuum throughout the Special Ventilation Zone with respect to the outside atmosphere.

e. Containment Vacuum Breaker System

The power operated valve in each vent line shall be tested during each refueling outage to demonstrate that a simulated containment vacuum of 0.5 psig will open the valve and a simulated accident signal will close the valve. The check and butterfly valves will be leak tested in accordance with specification 4.4.b during each refueling, except that the pressure will be applied in a direction opposite to that which would occur post-LOCA.

* See Note on Page TS 4.12-2.

Fluid Systems Vented (TS 4.4.a.4)

Venting of fluid systems which during post-accident conditions become an extension of the containment atmosphere is necessary to insure that possible leak paths of containment air in a post-accident situation will be verified as being leak tight or as needing repair. Those extensions of the containment atmosphere that are not vented prior to an ILRT include the following: RHR, SIS, ICS, CC, and SW. ILRT's shall be conducted in a manner as would occur had a containment isolation signal been initiated.

Isolating Leaks During the Test (TS 4.4.a.5)

Isolating excessive leak paths during a Type A test for later repair and completing the test ensures that the containment will be pressurized only once in conducting a Type A test. Type B or C leak testing paths that were isolated during a Type A test provides the "as found" leakage. Repairing and retesting the once isolated leak paths provides the "as left" leakage. Adding the pre-repair leakage to the ILRT results yields the "as found" total integrated leak rate while adding the post-repair leakage provides the "as left" total integrated leak rate.

Type A Test Acceptance Criterion (TS 4.4.a.6)

It has been recognized that the quality of the Containment Vessel and Penetration Seals used in the construction of the containment can permit meeting a 0.5 wt% per day leakage rate, ($L_a$). Assumptions for Containment Vessel leakage rate are provided in the UFSAR.[2] The acceptance criteria from Appendix J to 10CFR50, 0.75 $L_a$ or 0.375 wt%, is conservative. The assumptions used in the UFSAR conform to NRC Safety Guide 4 and result in offsite doses within the criteria set forth in 10CFR100 following the Design Basis Accident.

Type A Test Frequency (TS 4.4.a.7 and TS 4.4.a.8)

Integrated leak rate tests are done periodically to detect any deteriorating conditions that may adversely affect the ability of the primary reactor containment to perform its intended function. The Commission has determined that three tests at approximately equal intervals within ten years is a suitable frequency. 10CFR50, Appendix J explains Type A test schedule modifications applicable if an Integrated Leak Rate Test does not meet the acceptance criteria.

2. Containment Vessel Internal Spray System

A. System tests shall be performed once every operating cycle or once every 18 months, whichever occurs first. The test shall be performed with the isolation valves in the supply lines at the containment blocked closed.

B. The spray nozzles shall be checked for proper functioning at least every five years using either air with telltales or smoke tests to determine that all nozzles are clear.

C. The test will be considered satisfactory if control board indications or visual observations indicate all components have operated satisfactorily.

3. Containment Fan-Coil Units

Each fan-coil unit shall be tested once every operating cycle or once every 18 months, whichever occurs first, to verify proper operation of the motor-operated service water outlet valves and the fan coil emergency discharge and associated backdraft dampers.

b. Component Tests

1. Pumps

A. The safety injection pumps, residual heat removal pumps, and containment spray pumps shall be started and operated on recirculation flow quarterly during power operation and within one week after the plant is returned to power operation, if the test was not performed during plant shutdown.

B. Acceptable levels of performance shall be that the pumps start, reach their required developed head at miniflow, and operate for at least fifteen minutes on the miniflow line.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## RELATING TO AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. DPR-43

## WISCONSIN PUBLIC SERVICE CORPORATION
## WISCONSIN POWER AND LIGHT COMPANY
## MADISON GAS AND ELECTRIC COMPANY

## KEWAUNEE NUCLEAR POWER PLANT

## DOCKET NO. 50-305

### 1.0 INTRODUCTION

By letter dated February 19, 1990, Wisconsin Public Service Corporation (the licensee) requested an amendment to change the Technical Specifications (TSs) appended to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant (KNPP). The proposed amendment would change the TSs to delete TS 4.3 which concerns reactor coolant system (RCS) leak testing and weld examination requirements. These requirements are covered in TS 4.2.a, which requires that leak testing and inservice inspection be conducted in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

The proposed amendment also would revise TS 3.3.d.2 and TS 3.3.e.2 to eliminate the requirement to place the plant in a cold shutdown condition upon a long-term loss of one train of component cooling water (CCW) or service water (SW). This is to prevent having to place the plant in a mode which would demand the maximum support from the CCW or SW systems. By maintaining RCS temperature above 200°F but less than 350°F, the steam generators would be available as an additional means of decay heat removal.

The proposed amendment also includes several miscellaneous revisions that correct typographical errors, correct inconsistencies, and clarify the intent of certain technical specifications.

### 2.0 EVALUATION

The Westinghouse Standard Technical Specifications specify that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. Section 5.2.4 of the Standard Review Plan (NUREG-0800) states that inservice inspection programs are based on the general requirements of 10 CFR 50.55a, as detailed in Section XI of the ASME Code, "Rules for Inservice Inspection of Nuclear Power Plant Components." The current Kewaunee TS 4.2.a.1. also requires that inservice inspection of ASME Class 1, 2, and 3 components and supports be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. A redundant specification, TS 4.3, also covers RCS leak testing and weld examination requirements and inservice inspection requirements. TS 4.2.a.1 and TS 4.3 conflict on some of the specific requirements. The inconsistency creates ambiguity as to which specification predominates. Elimination of TS 4.3 will resolve this ambiguity and TS 4.2.a will solely govern test requirements of the reactor coolant system. Deletion of TS 4.3 is not a safety concern since implementation of an inservice inspection program in accordance with Section XI of the ASME Code is still required by TS 4.2.a.1. The proposed change is consistent with Standard Technical Specifications, 10 CFR 50.55a, and the Standard Review Plan and is, therefore, acceptable.

Concerning the proposed revision to TS 3.3.d.2 and TS 3.3.e.2 to replace the current action statement requiring cold shutdown within 36 hours with a new action statement, "Achieve and maintain the reactor coolant system $T_{avg}$ less than 350°F by use of alternate heat removal methods within an additional 36 hours;" this will prevent having to place the plant in a mode which would demand the maximum support from the CCW or SW systems when one train is out of service. Westinghouse Standard Technical Specifications require when operating with $T_{avg}$ greater than 350°F that when one ECCS subsystem is inoperable and cannot be restored within 72 hours the plant must be in Hot Standby within the next 6 hours and Hot Shutdown in the following 6 hours. Likewise, when $T_{avg}$ is less than 350°F, Standard Technical Specifications require a minimum of one ECCS subsystem to be maintained operable. With no ECCS subsystem operable because of the inoperability of the residual heat removal (RHR) heat exchanger or RHR pump, the RCS $T_{avg}$ is to be maintained less than 350°F by use of alternate heat removal methods. The Kewaunee Technical Specifications on RHR also do not require cold shutdown for the loss of one or more trains of RHR. This is due to the fact that the RHR system is the primary method of decay heat removal below 350°F. Since the CCW system cools the RHR heat exchangers and SW cools the CCW system, heat removal by means of RHR depends on the operability of both the CCW and SW systems.

The proposed change to TS 3.3.d.2 and 3.3.e.2, which currently require going to cold shutdown in the event of a long-term loss of one train of CCW or SW, would revise the action statements to maintain the plant above the cold shutdown mode (200°F-350°F) where additional heat removal methods are available via the steam generators. NUREG-1024, "Technical Specifications-Enhancing the Safety Impact," also recommends for CCW and SW system inoperability that it appears more appropriate that the plant be maintained in hot shutdown with the RCS between 200°F and 350°F where, in addition to having the one remaining RHR system loop, the RCS loops would also be available, any one of which would be able to remove decay heat. The proposed change is a precautionary measure that should reduce the probabliity of losing all auxiliary cooling below 350°F by allowing the plant to avoid relying solely upon one train of RHR for decay heat removal. Remaining above cold shutdown should be the preferred mode of operation from the standpoint of providing the least risk to the public. This portion of the proposed change is, therefore, acceptable.

The remaining changes described in the proposed amendment correct typographical errors, correct minor inconsistencies, and clarify the intent of certain technical specifications. The changes are necessary to maintain the accuracy of the Technical Specifications. Since these proposed changes are administrative in nature and do not change the intent of any technical specifications, they are not safety significant, and are, therefore, acceptable.

## 3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

## 5.0 CONCLUSION

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

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