ATTACHMENT 2

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To

Document Control Desk (NRC)

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PROPOSED TS AMENDMENT NO. 149

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## BASIS

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

A minimum shutdown margin of greater than or equal to 5% $\Delta k/k$ must be maintained in the core. A boron concentration of 2100 ppm, as required by TS 3.8.a.5, is sufficient to ensure an adequate margin of safety. The specification for REFUELING OPERATIONS shutdown margin is based on a dilution during refueling accident.[2] With an initial shutdown margin of 5% $\Delta k/k$, under the postulated accident conditions, it will take longer than 30 minutes for the reactor to go critical. This is ample time for the operator to recognize the audible high count rate signal, and isolate the reactor makeup water system. Periodic checks of refueling water boron concentration ensure that proper shutdown margin is maintained. Specification 3.8.a.6 allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

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

The one hundred hour decay time following plant shutdown is consistent with the assumption used in the dose calculation for the fuel handling accident. The requirement for the spent fuel pool sweep system, including charcoal adsorbers, to be operating when spent fuel movement is being made provides added assurance that the off-site doses will be within acceptable limits in the event of a fuel handling accident. The spent fuel pool sweep system is designed to sweep the atmosphere above the refueling pool and release to the Auxiliary Building vent during fuel handling operations. Normally, the charcoal adsorbers are bypassed but for purification operation, the bypass dampers are closed routing the air flow through the charcoal adsorbers. If the dampers

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[1]USAR Section 9.5.2

[2]USAR Section 14.1

do not close tightly, bypass leakage could exist to negate the usefulness of the charcoal adsorber. If the spent fuel pool sweep system is found not to be operating, fuel handling within the Auxiliary Building will be terminated until the system can be restored to the operating condition.

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

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

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

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

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

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

The presence of a licensed senior reactor operator at the site and designated in charge provides qualified supervision of the REFUELING OPERATIONS during changes in core geometry.[3]

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[3]USAR Section 13.2.1

## BASIS - Control Room Post-Accident Recirculation System (TS 3.12)

The Control Room Post-Accident Recirculation System is designed to filter the Control Room atmosphere during Control Room isolation conditions. The Control Room Post-Accident Recirculation System is designed to automatically start upon SIS or high radiation signal.

If the system is found to be inoperable, there is no immediate threat to the Control Room and reactor operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within 7 days, the reactor is placed in HOT STANDBY until the repairs are made.

Table TS 3.1-1
has been deleted

## TABLE TS 3.5-2

## INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP

| | | 1 | 2 | 3 | 4 | 5 | 6 |
|---|---|---|---|---|---|---|---|
| NO. | FUNCTIONAL UNIT | NO. OF CHANNELS | NO. OF CHANNELS TO TRIP | MINIMUM OPERABLE CHANNELS | MINIMUM DEGREE OF REDUNDANCY | PERMISSIBLE BYPASS CONDITIONS | OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET |
| 1 | Manual | 2 | 1 | 1 | - | | Maintain HOT SHUTDOWN |
| 2 | Nuclear Flux Power Range(1) | | | | | | |
| | Low setting | 4 | 2 | 3 | 1 | P-10 | Maintain HOT SHUTDOWN |
| | High setting | 4 | 2 | 3 | 1 | | |
| | Positive rate | 4 | 2 | 3 | 1 | | |
| | Negative rate | 4 | 2 | 3 | 1 | | |
| 3 | Nuclear Flux Intermediate Range | 2 | 1 | 1 | - | P-10 | Maintain HOT SHUTDOWN(3) |
| 4 | Nuclear Flux Source Range | 2 | 1 | 1 | - | P-6 | Maintain HOT SHUTDOWN(3) |
| 5 | Overtemperature $\Delta T$ | 4 | 2 | 3 | 1 | | Maintain HOT SHUTDOWN |
| 6 | Overpower $\Delta T$ | 4 | 2 | 3 | 1 | | Maintain HOT SHUTDOWN |
| 7 | Low Pressurizer Pressure | 4 | 2 | 3 | 1 | P-7 | Maintain HOT SHUTDOWN |
| 8 | High Pressurizer Pressure | 3 | 2 | 2 | - | | Maintain HOT SHUTDOWN |



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## TABLE TS 3.5-2

## INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP

**NOTES**

(1) One additional channel may be taken out of service for zero power physics testing.

(2) Deleted

(3) When a block condition exists, maintain normal operation.

(4) Underfrequency on the 4-kV buses trips the Reactor Coolant Pump breakers, which in turn trips the reactor when power is above P-7.

**P-6** 1 of 2 Intermediate Range Nuclear Instrument Channels indicates > $10^{-5}$% power

**P-7** 3 of 4 Power Range Nuclear Instrument channels < 10% power AND
2 of 2 Turbine Impulse Pressure Channels < 10% power

**P-8** 3 of 4 Power Range Nuclear Instrument Channels < 10% power

**P-10** 2 of 4 Power Range Nuclear Instrument Channels > 10% power

## 4.2 ASME CODE CLASS IN-SERVICE INSPECTION AND TESTING

### APPLICABILITY

Applies to in-service structural surveillance of the ASME Code Class components and supports and functional testing of pumps and valves.

### OBJECTIVE

To assure the continued integrity and operational readiness of ASME Code Class 1, 2, 3, and MC components.

### SPECIFICATION

a. ASME Code Class 1, 2, 3, and MC Components and Supports

1. In-service inspection of ASME Code Class 1, Class 2, Class 3, and Class MC components and supports shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g), except where relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). The testing and surveillance of shock suppressors (snubbers) is detailed in TS 3.14 and TS 4.14.

2. In-service testing of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(f), except where relief has been granted by the Commission pursuant to 10 CFR 50.55a(f)(6)(i).

3. Surveillance testing of pressure isolation valves:

    a. Periodic leakage testing[1] on each valve listed in Table TS 3.1-2 shall be accomplished prior to entering the OPERATING mode after every time the plant is placed in the COLD SHUTDOWN condition for refueling, after each time the plant is placed in a 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.

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[(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.

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.

b. Steam Generator Tubes

Examinations of the steam generator tubes shall be in accordance with the in-service inspection program described herein. The following terms are defmed to clarify the requirements of the inspection program.

Imperfection is an exception to the dimension, finish, or contour required by drawing or specification.

Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

% Degradation is an estimated % of the tube wall thickness affected or removed by degradation.

Degraded Tube means a tube contains an imperfection $\geq$ 20% of the nominal wall thickness caused by degradation.

Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

Tube Inspection means an inspection of the steam generator tube from the point of entry (e.g., hot leg side) completely around the U-bend to the top support of the opposite leg (cold leg).

Tube is the Reactor Coolant System pressure boundary past the hot leg side of the tubesheet and before the cold leg side of the tubesheet.

Plugged Tube is a tube intentionally removed from service by plugging in the hot and cold legs because it is defective, or because its continued integrity could not be assured.

Repaired Tube is a tube that has been modified to allow continued service consistent with plant Technical Specifications regarding allowable tube wall degradation, or to prevent further tube wall degradation. A tube without repairs is a nonrepaired tube. This definition does not apply to the portion of the tube below the F* or EF* distance provided the tube is not degraded (i.e., no detectable degradation permitted) within the F* distance for F* tubes and within the EF* distance for EF* tubes.

Laser Weld Repaired Sleeved Tube is a tube with a Westinghouse mechanical hybrid expansion joint sleeve that has been returned to operable status by use of a laser welded repair process.

F* Distance is the distanee of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube froin the tubesheet. The F* distance is equal to 1.12 inches (plus an allowance for NDE uncertainty) and is measured downward from the bottom of the uppermost roll transition. The F* distance applies to roll expanded regions below the midpoint of the tubesheet.

F* Tube is a tube with degradation below the F* distance, equal to or greater than 50%, and has no indications of degradation within the F* distance.

EF* Distance is the distance of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tubesheet. The EF* distance is equal to 1.44 inches (plus an allowance for NDE uncertainty) and is measured downward from the bottom of the uppermost roll transition. The EF* distance applies to roll expanded regions above the midpoint of the tubesheet.

EF* Tube is a tube with degradation below the EF* distance, equal to or greater than 50%, and has no degradation withim the EF* distance.

1. Steam Generator Sample Selection and Inspection

   The in-service inspection may be limited to one steam generator on a rotating schedule encompassing the number of tubes determined in TS 4.2.b.2.a provided the previous inspections indicated that the two steam generators are performing in a like manner.

2. Steam Generator Tube Sample Selection and Inspection

   The tubes selected for each in-service inspection shall:

   a. Include at least 3% of the total number of nonrepaired tubes, in both steam generators, and 20% of the total number of repaired tubes in both steam generators. The tubes selected for these inspections shall be selected on a random basis except as noted below and in TS 4.2.b.2.b.

      Indications left in service as a result of application of the tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future REFUELING outages.

   b. Concentrate the inspection by selection of at least 50% of the tubes to be inspected from critical areas where experience in similar plants with similar water chemistry indicates higher potential for degradation.

## BASIS

The plant was not specifically designed to meet the requirements of Section XI of the ASME Code; therefore, 100% compliance may not be feasible or practical. However, access for in-serviee inspection was considered during the design and modifications have been made where practical to make provisions for maximum access within the limits of the current plant design. Where practical, the inspection of ASME Code Class 1, Class 2, Class 3, and Class MC components is performed in accordance with Section XI of the ASME Code. If a code required inspection is impractical, a request for a deviation from the requirement is submitted to the Commission for approval.

The basis for surveillance testing of the Reactor Coolant System pressure isolation valves identified in Table TS 3.1-2 is contained within "Order for Modification of License" dated April 20, 1981.

Technical Specification 4.2.b

These Technical Specifications provide the inspection and repair/plugging requirements for the steam generator tubes at the Kewaunee Nuclear Power Plant. Fulfilling these specifications will assure the KNPP steam generator tubes are inspected and maintained in a manner consistent with current NRC regulations and guidelines including the General Design Criteria in 10 CFR Part 50, Appendix A.

General Design Criterion (GDC) 14 "Reactor Coolant Pressure Boundary," and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," require that the reactor coolant pressure boundary have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Also, GDC 15, "Reactor Coolant System Design," requires that the Reactor Coolant System and associated auxiliary, control, and protection systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. Furthermore, GDC 32 "Inspection of Reactor Coolant System Pressure Boundary," requires that components that are part of the reactor coolant pressure boundary be designed to permit periodic inspection and testing of critical areas to assess their structural and leak tight integrity.

The NRC has developed guidance for steam generator tube inspections and maintenance including Regulatory Guides 1.83 and 1.121. Regulatory Guide 1.83, "In-service Inspection of Pressurized Water Reactor Steam Generator Tubes," forms the basis for many of the requirements in this section and should be consulted prior to any revisions. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," defines the minimum wall thickness in a steam generator tube, and may be applied to tube sleeves in determining their minimum wall thickness.

## 4.5 EMERGENCY CORE COOLING SYSTEM AND CONTAINMENT AIR COOLING SYSTEM TESTS

### APPLICABILITY

Applies to testing of the Emergency Core Cooling System and the Containment Air Cooling System.

### OBJECTIVE

To verify that the subject systems will respond promptly and perform their design functions, if required.

### SPECIFICATION

a. System Tests

1. Safety Injection System

A. System tests shall be performed once per operating cycle or once every 18 months, whichever occurs first. With the Reactor Coolant System pressure $\leq$ 350 psig and temperature $\leq$ 350°F, a test safety injection signal will be applied to initiate operation of the system.

B. The test will be considered satisfactory if control board indication or visual observations indicate that all components have received the safety injection signal in the proper sequence and timing. That is, the appropriate pump motor breakers shall have opened and closed, and all valves shall have completed their travel.

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. Verify a minimum of 76 spray nozzles per train are functioning properly by using an air or smoke test at a test interval not to exceed 10 years.

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

3. Containment Fancoil Units

Each fancoil 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 fancoil 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 quarterly during power operation and within 1 week after the plant is returned to power operation, if the test was not performed during plant shutdown.

B. Acceptable levels of performance are demonstrated by the pumps' ability to start and develop head within an acceptable range.

2. Valves

A. The containment sump outlet valves shall be tested during the pump tests.

B. The accumulator check valves shall be checked for OPERABILITY during each major REFUELING outage. The accumulator block valves shall be checked to assure "valve open" requirements during each major REFUELING outage.

C. Deleted

D. Spray additive tank valves shall be tested during each major REFUELING outage.

E. Deleted

F. Residual Heat Removal System valve interlocks shall be tested once per operating cycle.

## 4.13 RADIOACTIVE MATERIALS SOURCES

### APPLICABILITY

Applies to the possession, leak test, and record requirements for radioactive material sources required for operation of the facility.

### OBJECTIVE

To ensure that radioactive material sources which are beneficial to facility operation are available to the facility and these sources are verified to be free from leakage.

### SPECIFICATION

a. Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or the State.

b. Sources which contain by-product material that exceeds the quantities listed in 10 CFR 30.71, Schedule B, and all other sources containing > 0.1 microcuries shall be leak tested in accordance with this TS.

c. Any source specified by TS 4.13.b which is determined to be leaking shall be immediately withdrawn from use, repaired or disposed of in accordance with the Commission's regulations. Leaking is defined as the presence of .005 microcuries of the source's radioactive material on the test sample.

d. Each sealed source with a half-life > 30 days, and in any form other than gas, shall be tested for leakage at intervals not to exceed 6 months, except for:

   1. Startup sources inserted in the reactor vessel,
   2. Fission detectors following exposure to core flux,
   3. Irradiation sample sources inserted in the reactor vessel,
   4. Sources enclosed within the Eberline Model 1000 Multi-Source Gamma Calibrator,
   5. Sources enclosed within the Shepherd Model 89-400 Self-Contained Calibrator, and
   6. Hydrogen-3 sources.

e. Sources specified by TS 4.13.b which are in storage and not being used are exempt from the testing of TS 4.13.d. Prior to use or transfer to another licensee of such a source, the leakage test of TS 4.13.d shall be current.

f. Startup sources and fission detectors shall be leak tested prior to initial insertion into the reactor vessel or prior to being subjected to core flux.

## 4.17 CONTROL ROOM POSTACCIDENT RECIRCULATION SYSTEM

### APPLICABILITY

Applies to testing and surveillance requirements for the Control Room Postaccident Recirculation System in TS 3.12.

### OBJECTIVE

To verify the performance capability of the Control Room Postaccident Recirculation System.

### SPECIFICATION

a. At least onee per operating cycle or once every 18 months, whichever occurs first, the following conditions shall be demonstrated:

   1. Pressure drop across the combined HEFA filters and charcoal adsorber banks is < 6 inches of water and the pressure drop across any HEPA bank is < 4 inches of water at the system design flow rate (± 10%).

   2. Automatic initiation of the system on a high radiation signal and a safety injection signal.

b. 1. The in-place DOP test for HEPA filters shall be performed (1) at least once per 18 months and (2) after each complete or partial replacement of a HEPA filter bank or after any maintenance on the system that could affect the HEPA bank bypass leakage.

   2. The laboratory tests for activated carbon in the charcoal filters shall be performed (1) at least once per 18 months for filters in a standby status or after 720 hours of filter operation, and (2) following painting, fire, or chemical release in any ventilation zone communicating with the system.

   3. Halogenated hydrocarbon testing shall be performed after each complete or partial replaeement of a charcoal adsorber bank or after any maintenance on the system that could affect the charcoal adsorber bank bypass leakage.

   4. Each train shall be operated at least 10 hours each month.