ATTACHMENT 2

To

Proposed Amendment No. 105

Letter from C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Affected TS Pages

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3.1 REACTOR COOLANT SYSTEM

APPLICABILITY

Applies to the Operating status of the Reactor Coolant System (RCS).

OBJECTIVE

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

SPECIFICATIONS

a. Operational Components

1. Reactor Coolant Pumps

   A. At least one reactor coolant pump or one residual heat removal pump shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

   B. When the reactor is in the OPERATING mode, except for low power tests, both reactor coolant pumps shall be in operation.

2. Decay Heat Removal Capability

   A. At least TWO of the following FOUR heat sinks shall be operable whenever the average reactor coolant temperature is $\leq$ 350°F but > 200°F.

      1. Steam Generator 1A
      2. Steam Generator 1B
      3. Residual Heat Removal Train A
      4. Residual Heat Removal Train B

      If less than the above number of required heat sinks are operable, corrective action shall be taken immediately to restore the minimum number to the operable status.

B. TWO residual heat removal trains shall be operable whenever the average reactor coolant temperature is ≤ 200°F and irradiated fuel is in the reactor, except when in the REFUELING mode one train may be inoperable for maintenance.

1. Each residual heat removal train shall be comprised of:

    a) ONE operable residual heat removal pump

    b) ONE operable residual heat removal heat exchanger

    c) An operable flow path consisting of all valves and piping associated with the above train of components and required to remove decay heat from the core during normal shutdown situations. This flow path shall be capable of taking suction from the appropriate Reactor Coolant System hot leg and returning to the Reactor Coolant System.

2. If one residual heat removal train is inoperable, corrective action shall be taken immediately to return it to the operable status.

3. Pressurizer Safety Valves

A. At least one pressurizer safety valve shall be operable whenever the reactor head is on the reactor pressure vessel, except for a hydro test of the RCS the pressurizer safety valves may be blanked provided the power-operated relief valves and the safety valve on the discharge of the charging pump are set for test pressure plus 35 psi to protect the system.

B. Both pressurizer safety valves shall be operable whenever the reactor is critical.

4. Pressure Isolation Valves

A. All pressure isolation valves listed in Table TS 3.1-2 shall be functional as a pressure isolation device during OPERATING and HOT STANDBY modes, except as specified in 3.1.a.4.B. Valve leakage shall not exceed the amounts indicated.

B. In the event that integrity of any pressure isolation valve as specified in Table TS 3.1-2 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a non-functional valve are in, and remain in, the mode corresponding to the isolated condition.[1]

C. If TS 3.1.a.4.A and TS 3.1.a.4.B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the HOT SHUTDOWN condition within the next 4 hours, the INTERMEDIATE SHUTDOWN condition in the next 6 hours and the COLD SHUTDOWN condition within the next 24 hours.

5. Pressurizer Power-Operated Relief Valves (PORV) and PORV Block Valves

A. Two PORVs and their associated block valves shall be operable during HOT STANDBY and OPERATING modes.

1. If a pressurizer PORV is inoperable, the PORV shall be restored to an operable condition within one hour or the associated block valve shall be closed and maintained closed by administrative procedures to prevent inadvertent opening.

2. If a PORV block valve is inoperable, the block valve shall be restored to an operable condition within one hour or the block valve shall be closed with power removed from the valve; otherwise the unit shall be placed in the HOT SHUTDOWN condition using normal operating procedures.

6. Pressurizer Heaters

A. At least one group of pressurizer heaters shall have an emergency power supply available when the average RCS temperature is > 350°F.

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[1]Manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position with their power breakers locked out.

7. Reactor Coolant Vent System

A. A reactor coolant vent path from both the reactor vessel head and pressurizer steam space shall be operable and closed prior to the average RCS temperature being heated > 200°F except as specified in TS 3.1.a.7.B and TS 3.1.a.7.C below.

B. When the average RCS temperature is > 200°F, any one of the following conditions of inoperability may exist:

1. Both of the parallel vent valves in the reactor vessel vent path are inoperable.

2. Both of the parallel vent valves in the pressurizer vent path are inoperable.

If operability is not restored within 30 days, then within one hour action shall be initiated to:

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

C. If no Reactor Coolant System vent paths are operable, restore at least one vent path to operable status within 72 hours. If operability is not restored within 72 hours, then within 1 hour action shall be initiated to:

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

b. Heatup and Cooldown Limit Curves for Normal Operation

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS 3.1-1 and TS 3.1-2 for the service period up to 15 equivalent full-power years.

    A. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

    B. Figures TS 3.1-1 and TS 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. The secondary side of the steam generator must not be pressurized > 200 psig if the temperature of the steam generator is < 70°F.

3. The pressurizer cooldown and heatup rates shall not exceed 200°F/hr and 100°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is > 320°F.

c. Maximum Coolant Activity

The total specific activity of the reactor coolant due to nuclides with half-lives of more than 30 minutes, excluding tritium, shall not exceed

$$A=\frac{91}{\bar{E}} \qquad \frac{\mu Ci}{cc}$$

whenever the reactor is critical or the average temperature is > 500°F ($\bar{E}$ is the average sum of the beta and gamma energies in Mev per disintegration).

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d. Leakage of Reactor Coolant

1. Any Reactor Coolant System leakage indication in excess of 1 gpm shall be the subject of an investigation and evaluation initiated within 4 hours of the indication. Any indicated leak shall be considered to be a real leak until it is determined that no unsafe condition exists. If the Reactor Coolant System leakage exceeds 1 gpm and the source of leakage is not identified within 12 hours, the reactor shall be placed in the HOT SHUTDOWN condition utilizing normal operating procedures. If the source of leakage exceeds 1 gpm and is not identified within 48 hours, the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.

2. Reactor coolant-to-secondary leakage through the steam generator tubes shall be limited to 500 gallons per day through any one steam generator except if the tube support plate alternate plugging criteria is applied. If the alternate plugging criteria is applied then leakage shall be limited to 150 gallons per day through any one steam generator. With tube leakage greater than the above limit, reduce the leakage rate within 4 hours or be in COLD SHUTDOWN within the next 36 hours.

3. If the sources of leakage other than that in 3.1.d.2 have been identified and it is evaluated that continued operation is safe, operation of the reactor with a total Reactor Coolant System leakage rate not exceeding 10 gpm shall be permitted. If leakage exceeds 10 gpm, the reactor shall be placed in the HOT SHUTDOWN condition within 12 hours utilizing normal operating procedures. If the leakage exceeds 10 gpm for 24 hours, the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.

4. If any reactor coolant leakage exists through a non-isolable fault in a Reactor Coolant System component (exterior wall of the reactor vessel, piping, valve body, relief valve leaks, pressurizer, steam generator head, or pump seal leakoff), the reactor shall be shut down; and cooldown to the COLD SHUTDOWN condition shall be initiated within 24 hours of detection.

5. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be in operation with one of the two systems sensitive to radioactivity. Either system may be out of operation for up to 12 hours provided at least one system is operable.

e. Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration

1. Concentrations of contaminants in the reactor coolant shall not exceed the following limits when the reactor coolant temperature is > 250°F.

CONTAMINANT	NORMAL STEADY-STATE OPERATION (ppm)	TRANSIENT LIMITS (ppm)
A. Oxygen	0.10	1.00
B. Chloride	0.15	1.50
C. Fluoride	0.15	1.50

2. If any of the normal steady-state operating limits as specified in TS 3.1.e.1 above are exceeded, or if it is anticipated that they may be exceeded, corrective action shall be taken immediately.

3. If the concentrations of any of the contaminants cannot be controlled within the transient limits of TS 3.1.e.1 above or returned to the normal steady-state limit within 24 hours, the reactor shall be brought to the COLD SHUTDOWN condition, utilizing normal operating procedures, and the cause shall be ascertained and corrected. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values; otherwise a safety review by the Plant Operations Review Committee shall be made before starting.

4. Concentrations of contaminants in the reactor coolant shall not exceed the following maximum limits when the reactor coolant temperature is ≤ 250°F.

CONTAMINANT	NORMAL CONCENTRATION (ppm)	TRANSIENT LIMITS (ppm)
A. Oxygen	Saturated	Saturated
B. Chloride	0.15	1.50
C. Fluoride	0.15	1.50

5. If the transient limits of TS 3.1.e.4 are exceeded or the concentrations cannot be returned to normal values within 48 hours, the reactor shall be brought to the COLD SHUTDOWN condition and the cause shall be ascertained and corrected.

6. To meet TS 3.1.e.1 and TS 3.1.e.4 above, reactor coolant pump operation shall be permitted for short periods, provided the coolant temperature does not exceed 250°F.

f. Minimum Conditions for Criticality

1. Except during low-power physics tests, the reactor shall not be made critical unless the moderator temperature coefficient is negative.

2. The reactor shall not be brought to a critical condition until the pressure-temperature state is to the right of the criticality limit line shown in Figure TS 3.1-1.

3. Except during low-power physics tests, when the reactor coolant temperature is in a range where the moderator temperature coefficient is positive, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

4. The reactor shall be maintained subcritical by at least 1% $\Delta k/k$ until normal water level is established in the pressurizer.

BASES - Operational Components (TS 3.1.a)

Reactor Coolant Pumps (TS 3.1.a.1)

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

Part 1 of the specification requires that both reactor coolant pumps be operating when the reactor is in power operation to provide core cooling. Planned power operation with one loop out of service is not allowed in the present design because the system does not meet the single failure (locked rotor) criteria requirement for this mode of operation. The flow provided in each case in Part 1 will keep DNBR well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. One pump operation is not permitted except for tests. Upon loss of one pump below 10% full power, the core power shall be reduced to a level below the maximum power determined for zero power testing. Natural circulation can remove decay heat up to 10% power. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost.[2]

Decay Heat Removal Capabilities (TS 3.1.a.2)

When the average reactor coolant temperature is ≤ 350°F a combination of the available heat sinks is sufficient to remove the decay heat and provide the necessary redundancy to meet the single failure criterion.

When the average reactor coolant temperature is ≤ 200°F, the plant is in a COLD SHUTDOWN condition and there is a negligible amount of sensible heat energy stored in the Reactor Coolant System. Should one residual heat removal train become inoperable under these conditions, the remaining train is capable of removing all of the decay heat being generated.

Pressurizer Safety Valves (TS 3.1.a.3)

Each of the pressurizer safety valves is designed to relieve 325,000 lbs. per hour of saturated steam at its setpoint. Below 350°F and 350 psig, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate protection against overpressurization.

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[2] USAR Section 7.2.2

Pressure Isolation Valves (TS 3.1.a.4)

The Basis for the Pressure Isolation Valves is discussed in the Reactor Safety Study (RSS), WASH-1400, and identifies an intersystem loss-of-coolant accident in a PWR which is a significant contributor to risk from core melt accidents (EVENT V). The design examined in the RSS contained two in-series check valves isolating the high pressure Primary Coolant System from the Low Pressure Injection System (LPIS) piping. The scenario which leads to the EVENT V accident is initiated by the failure of these check valves to function as a pressure isolation barrier. This causes an overpressurization and rupture of the LPIS low pressure piping which results in a LOCA that bypasses containment.(3)

PORVs and PORV Block Valves (TS 3.1.a.5)

The pressurizer power-operated relief valves (PORVs) operate as part of the pressurizer pressure control system. They are intended to relieve RCS pressure below the setting of the code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a PORV become inoperable.

Pressurizer Heaters (TS 3.1.a.6)

Pressurizer heaters are vital elements in the operation of the pressurizer which is necessary to maintain system pressure. Loss of energy to the heaters would result in the inability to maintain system pressure via heat addition to the pressurizer. Hot functional tests(4) have indicated that one group of heaters is required to overcome ambient heat losses. Placing heaters necessary to overcome ambient heat losses on emergency power will assure the ability to maintain pressurizer pressure. Annual surveillance tests are performed to ensure heater operability.

Reactor Coolant Vent System (TS 3.1.a.7)

The function of the high point vent system is to vent noncondensible gases from the high points of the RCS to assure that core cooling during natural circulation will not be inhibited. The operability of at least one vent path from both the reactor vessel head and pressurizer steam space ensures the capability exists to perform this function.

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(3) Order for Modification of License dated 4/20/81

(4) Hot functional test (PT-RC-31)

The vent path from the reactor vessel head and the vent path from the pressurizer each contain two independently emergency powered, energize to open, valves in parallel and connect to a common header that discharges either to the containment atmosphere or to the pressurizer relief tank. The lines to the containment atmosphere and pressurizer relief tank each contain an independently emergency powered, energize to open, isolation valve. This redundancy provides protection from the failure of a single vent path valve rendering an entire vent path inoperable.

A flow restriction orifice in each vent path limits the flow from an inadvertent actuation of the vent system to less than the flow capacity of one charging pump.[5]

Heatup and Cooldown Limit Curves for Normal Operation (TS 3.1.b)

Fracture Toughness Properties - (TS 3.1.b.1)

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the ASME Boiler and Pressure Vessel Code[6], and the calculation methods of Footnote[7]. The postirradiation fracture toughness properties of the reactor vessel belt line material were obtained directly from the Kewaunee Reactor Vessel Material Surveillance Program.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Nonmandatory Appendix G in Section Ill of the ASME Boiler and Pressure Vessel Code, and are discussed in detail in Footnote[8].

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[5] Letter from E. R. Mathews to S. A. Varga dated 5/21/82

[6] ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components" Section III, Summer 1984 Addenda, Non-Mandatory Appendix G - "Protection Against Non-ductile Failure."

[7] Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques, ASTM designation E262-70, 1975 Book of ASTM Standards, Part 45, pp. 756-763.

[8] P. K. Nair and E. B. Norris, "Pressure/Temperature Operating Curves and Assessment of RT_{PTS} Concerns for Kewaunee Nuclear Plant," SWRI Project 06-8919, April, 1986.

The method specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by the pressure gradient. Thus, the governing equation for the heatup-cooldown analysis is:

$$2\ K_{Im} + K_{It} \leq K_{IR} \qquad (3.1b\text{-}1)$$

where

K_{Im} is the stress intensity factor caused by membrane (pressure) stress

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IR} is provided by the Code as a function of temperature relative to the RT_{NDT} of the material.

From equation (3.1b-1) the variables that affect the heatup and cooldown analysis can be readily identified. K_{Im} is the stress intensity factor due to membrane (pressure) stress. K_{It} is the thermal (bending) stress intensity factor and accounts for the linearly varying stress in the vessel wall due to thermal gradients. During heatup K_{It} is negative on the inside and positive on the outer surface of the vessel wall. The signs are reversed for cooldown and, therefore, an ID or an OD one quarter thickness surface flaw is postulated in whichever location is more limiting. K_{IR} is dependent on irradiation and temperature and, therefore, the fluence profile through the reactor vessel wall and the rates of heatup and cooldown are important. Details of the procedure used to account for these variables is explained in the following text.

Following the generation of pressure-temperature curves for both the steady-state (zero rate of change of temperature) and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the OD to the ID location. The pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup with the exception that the controlling location is always at the ID. The thermal gradients induced during cooldown tend to produce tensile stresses at the ID location and compressive stresses at the OD position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher K_{IR} for finite cooldown rates than for steady-state under certain conditions.

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed above. The derivation of the limit curves is consistent with NRC Regulatory Standard Review Plan Directorate of Licensing, Section 5.3.2, "Pressure-Temperature Limits" 1974 and Footnote[9].

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. As presented in WCAP 9878[10], weld metal Charpy test specimens from Capsule R indicate that the core region weld metal exhibits the largest shift in RT_{NDT} (235°F).

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[9] ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components" Section III, Summer 1984 Addenda, Non-Mandatory Appendix G - "Protection Against Non-ductile Failure."

[10] S.E. Yanichko, et al, "Analysis of Capsule R from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 9878, March, 1981.

The results of Irradiation Capsules V and R analyses are presented in WCAP 8908[11] and WCAP 9878, respectively. Heatup and cooldown limit curves for normal operation of the reactor vessel are presented in Figures TS 3.1-1 and TS 3.1-2[12] and represent an operational time period of 15 effective full-power years.

Pressurizer Limits - (TS 3.1.b.3)

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with Code requirements. In-plant testing and calculations have shown that a pressurizer heatup rate of 100°F/hr cannot be achieved with the installed equipment.

Maximum Coolant Activity (TS 3.1.c)

This specification is based on the evaluation of the consequences of a postulated rupture of a steam generator tube when the maximum activity in the reactor coolant is at the allowable limit. The potential release of activity to the atmosphere has been evaluated to insure that the public is protected.

Rupture of a steam generator tube would allow reactor coolant activity to enter the secondary system. The major portion of this activity is noble gases[13] which would be released to the atmosphere from the air ejector or a relief valve. Activity could continue to be released until the operator could reduce the Reactor Coolant System pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single tube, followed by isolation of the faulty steam generator by the operator within one-half hour after the event. During this period, 120,000 lbs. of reactor coolant are discharged into the steam generator.[13]

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(11) S. E. Yanichko, S. L. Anderson, and K. V. Scott, "Analysis of Capsule V from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 8908, January 1977.

(12) Letter from P. S. VanTeslaar (Westinghouse) to C. W. Giesler (WPSC) dated April 30, 1981, transmitting KNPP heatup and cooldown curves based on Capsule R results.

(13) USAR Section 14.2.4

The limiting off-site dose is the whole-body dose resulting from immersion in the cloud containing the released activity. Radiation would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, for purposes of analysis, the simple model of a semi-infinite cloud, which gives an upper limit to the potential gamma dose, has been used. The semi-infinite cloud model is applicable to the beta dose because of the short range of beta radiation in air. The effectiveness of clothing as shielding against beta radiation is neglected and therefore the analysis model also gives an upper limit to the potential beta dose.

The combined gamma and beta dose from a semi-infinite cloud is given by:

$$Dose,\ rem = 1/2\ [\bar{E} \cdot A \cdot V \cdot \frac{X}{Q} \cdot (3.7 \times 10^{10})\ \ (1.33 \times 10^{-11})]$$

Where: $\bar{E}$ = average energy of betas and gammas per disintegration (Mev/dis)

A = primary coolant activity (Ci/m^3)

$\bar{E}A$ = 91 Mev Ci/dis m^3 (the maximum per this specification)

$\frac{X}{Q}$ = 2.9×10^{-4} sec/m^3, the 0-2 hr. dispersion coefficient at the site boundary prescribed by the Commission

V = 77 m^3, which corresponds to a reactor coolant liquid mass of 120,000 lbs.

The resultant dose is < 0.5 rem at the site boundary.

Leakage of Reactor Coolant (TS 3.1.d.1)[14]

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage would not exceed the limits of 10 CFR Part 20. This is shown as follows:

If the reactor coolant activity is $91/\bar{E}$ μCi/cc ($\bar{E}$ = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling System vent line, the yearly whole body dose resulting from this activity at the site boundary, using an annual average X/Q = 2.0×10^{-6} sec/m^3, is 0.09 rem/yr, compared with the 10 CFR Part 20 limits of 0.5 rem/yr.

With the limiting reactor coolant activity and assuming initiation of a 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet header would annunciate in the control room and initiate closure of the vent line from the surge tank in the Component Cooling System, within less than 1 minute. In the case of failure of the closure of the vent line and resulting continuous discharge to the atmosphere via the component cooling surge tank vent, the resultant dose rate at the site boundary would be 0.09 rem/yr as given above.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation of 1 gpm for an unidentified source of leakage is sufficiently above the minimum detectable leak rate to provide a reliable indication of leakage, and is well below the capacity of one charging pump (60 gpm).

Twelve hours of operation before placing the reactor in the HOT SHUTDOWN condition are required to provide adequate time for determining whether the leak is into the containment or into one of the closed systems and to identify the leakage source.

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[14] USAR Sections 6.5, 11.2.3, 14.2.4

TS 3.1.d.2

The 150 gpd leakage limit through any one steam generator (300 gpd total) is specified to ensure tube integrity is maintained in the event of a main steam line break or under loss-of-coolant accident conditions if the tube support plate alternate plugging criteria is applied. By maintaining an operating leakage limit of 150 gpd per steam generator, the total primary-to-secondary leak rate following a main steam line break is limited to < 260 gpm in the faulted loop and 150 gpd for the intact loop. This will limit off-site doses to within a small fraction of the 10 CFR Part 100 guidelines.

The operating leakage limit of 300 gpd total (150 gpd per steam generator) is based on a Kewaunee site specific radiological evaluation performed to support an alternate plugging criteria for the steam generators. The steam generator plugging criteria is specified in TS 4.2.b.

TS 3.1.d.3

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the plant operating staff and will be documented in writing and approved by either the Plant Manager or his designated alternate. Under these conditions, an allowable Reactor Coolant System leak rate of 10 gpm has been established. This explained leak rate of 10 gpm is within the capacity of one charging pump as well as being equal to the capacity of the Steam Generator Blowdown Treatment System.

TS 3.1.d.4

The provision pertaining to a non-isolable fault in a Reactor Coolant System component is not intended to cover steam generator tube leaks, valve bonnets, packings, instrument fittings, or similar primary system boundaries not indicative of major component exterior wall leakage.

TS 3.1.d.5

If leakage is to the containment, it may be identified by one or more of the following methods:

A. The containment air particulate monitor is sensitive to low leak rates. The rates of reactor coolant leakage to which the instrument is sensitive is dependent upon the presence of corrosion product activity.

B. The containment radiogas monitor is less sensitive and is used as a backup to the air particulate monitor. The sensitivity range of the instrument is approximately 2 gpm to > 10 gpm.

C. Humidity detection provides a backup to A. and B. The sensitivity range of the instrumentation is from approximately 2 gpm to 10 gpm.

D. A leakage detection system is provided which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by fancoils of the Containment Air Cooling System and thus provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. The fancoil units drain to the containment sump, and all leakage collected by the containment sump will be pumped to the waste holdup tank. Pump running time will be monitored in the control room to indicate the quantity of leakage accumulated.

If leakage is to another closed system, it will be detected by the area and process radiation monitors and/or inventory control.

Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration (TS 3.1.e)

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in TS 3.1.e.1 and TS 3.1.e.4, the integrity of the Reactor Coolant System is assured under all operating conditions.[15]

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank[16]. Because of the time-dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus, the time periods for corrective action to restore concentrations within the limits have been established. If the corrective action has not been effective at the end of the time period, reactor cooldown will be initiated and corrective action will continue.

The effects of contaminants in the reactor coolant are temperature dependent. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values; otherwise a safety review by the Plant Operations Review Committee is required before startup.

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(15) USAR Section 4.2

(16) USAR Section 9.2

Minimum Conditions for Criticality (TS 3.1.f)

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range. The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is greatest. Later in the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients either will be less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range.[17][18]

Suitable physics measurements of moderator coefficients of reactivity will be made as part of the startup testing program to verify analytical predictions.

The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operation, as a result of either an increase in moderator temperature or a decrease in coolant pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[19] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction in moderator density.

The requirement that the reactor is not to be made critical except as specified in TS 3.1.f.2 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil-ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by the pressurizer heaters.

The shutdown margin specified in TS 3.10 precludes the possibility of accidental criticality as a result of an increase in moderator temperature or a decrease in coolant pressure.[20]

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(17) USAR Table 3.2-1

(18) USAR Figure 3.2-8

(19) USAR Figure 3.2-9

(20) USAR Table 3.2-1

The requirement that the pressurizer is partly voided when the reactor is < 1% subcritical assures that the Reactor Coolant System will not be solid when criticality is achieved.

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 and 3 components.

SPECIFICATION

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

1. In-service inspection of ASME Code Class 1, Class 2 and Class 3 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 Sections 3.14 and 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(g), except where relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(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 defined 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 or repair 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). For a tube in which the tube support plate alternate plugging criteria has been applied, the inspection will include all the hot leg and all the cold leg tube support plate intersections.

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.

1. <u>Steam Generator Sample Selection and Inspection</u>

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. <u>Steam Generator Tube Sample Selection and Inspection</u>

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 3% 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 in 4.2.b.2.b.

However, if the tube support plate plugging criteria is applied, then the following inspections are required:

1. A 100% bobbin probe inspection of the tube support plate intersections.

2. An inspection using a rotating pancake coil (RPC) probe of all indications at tube support plates > 1.5 volts bobbin coil signal amplitude. Once an indication is characterized as ODSCC, RPC inspection at alternate refueling outages is acceptable.

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.

c. Include the inspection of all non-plugged tubes which previous inspections revealed in excess of 20% degradation. The previously degraded tubes need only be inspected about the area of previous degradation indication if their inspection is not employed to satisfy 4.2.b.2.a and 4.2.b.2.b above.

d. The second and third sample inspections during each in-service inspection may be less than the full length of each tube by concentrating the inspection on those areas of the tubesheet array and on those portions of the tubes where tubes with imperfections were previously found.

e. If a tube does not permit the passage of the eddy current inspection probe the entire length and through the U-bend, this shall be recorded and an adjacent tube shall be inspected. The tube which did not allow passage of the eddy current probe shall be considered degraded.

The results of each sample inspection shall be classified into one of the following three categories, and actions taken as described in Table 4.2-2.

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes, and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

NOTE: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

3. Inspection Frequencies

The above required in-service inspections of steam generator tubes shall be performed at the following frequencies:

a. In-service inspections shall be performed at refueling intervals not more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category; or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the in-service inspection of a steam generator conducted in accordance with Table 4.2-2 fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 4.2.b.3.a and the interval can be extended to a 40-month period.

c. Additional, unscheduled in-service inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.2-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tubesheet welds) in excess of the limits of TS 3.1.d and TS 3.4.a.4, or

   2. A seismic occurrence greater than the Operating Basis Earthquake, or

   3. A loss-of-coolant accident requiring actuation of the engineering safeguards, where the cooldown rate of the Reactor Coolant System exceeded 100°F/hr, or

   4. A main steam line or feedwater line break, where the cooldown rate of the Reactor Coolant System exceeded 100°F/hr.

d. If the type of steam generator chemistry treatment is changed significantly, the steam generators shall be inspected at the next outage of sufficient duration following 3 months of power operation since the change.

e. Tubes for which the tube support plate alternate plugging criteria has been applied shall be inspected during all future refueling outages.

4. Plugging or Repair Limit Criteria

The following criteria apply independently to tube and sleeve wall degradation, except as specified in TS 4.2.b.5 for the tube support plate alternative plugging criteria.

a. Any tube which, upon inspection, exhibits tube wall degradation of 50% or more shall be plugged or repaired prior to returning the steam generator to service. If significant general tube thinning occurs, this criterion will be reduced to 40% wall degradation. Repair methods will be submitted under 10 CFR 50.90 to be incorporated as an amendment in the facility license. The Commission will review the repair method, issue a significant hazards determination, and amend the facility license.

b. Any sleeve which, upon inspection, exhibits wall degradation of 31% or more shall be plugged prior to returning the steam generator to service. Figure 4.2-1 illustrates the application of tube, sleeve, and tube/sleeve joint plugging limit criteria.

5. Tube Support Plate Alternate Plugging Criteria

The following criteria applies to indications within the thickness of the tube support plate intersections.

a. A tube may remain in service if the bobbin coil signal amplitude of the indication is ≤ 3.5 volts except as provided by TS 4.2.b.5.b, regardless of the depth of the wall penetration. This is provided that the projected end-of-cycle distribution of indications is verified to result in a primary-to-secondary leakage of < 260 gpm in the faulted loop during a postulated main steam line event.

b. All bobbin coil signal amplitudes > 1.5 volts shall be inspected with a RPC probe. The RPC results shall be evaluated to support ODSCC as the dominant degradation mechanism. Tubes with RPC indications not attributable to ODSCC or with circumferential indications shall be evaluated for plugging in accordance with TS 4.2.b.4.

c. A tube shall be plugged or repaired if the bobbin coil signal amplitude of the indication is > 3.5 volts.

6. Reports

a. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or repairing, the number of tubes plugged or repaired shall be reported to the Commission within 30 days.

b. The results of the steam generator tube in-service inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of a degradation.

   3. Identification of tubes plugged.

   4. Identification of tubes repaired.

c. Results of a steam generator tube inspection which fall into Category C-3 require prompt (within 4 hours) notification of the Commission consistent with 10 CFR 50.72(b)(2)(i). A written follow up report shall be submitted to the Commission consistent with TS 4.2.b.6.a, using the Licensee Event Report System to satisfy the intent of 10 CFR 50.73(a)(2)(ii).

d. The results of inspections for tubes which the tube support plate alternate plugging criteria has been applied shall be reported to the Commission within 30 days consistent with TS 4.2.6.a. The report shall include:

   1. Listing of applicable tubes.

   2. Location (applicable intersections per tube) and bobbin coil voltage amplitude.

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-service 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 and Class 3 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.

Steam Generator Tubes (TS 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.

In 1992, an alternate plugging criteria was approved for tubes experiencing ODSCC within the thickness of the tube support plates. The alternate plugging criteria is based on bobbin coil voltage amplitude. These eddy current measurements are directly correlated to tube integrity issues including tube burst margins and potential for tube leakage under postulated accident conditions. Regulatory Guide 1.121 and the GDCs are satisfied by the alternate tube plugging criteria as shown in WCAP-12985 "Kewaunee Steam Generator Tube Plugging Criteria for ODSCC at Tube Support Plates."

<u>Steam Generator Sample Selection and Inspection (TS 4.2.b.1)</u>

If the steam generators are shown to be performing in a like manner, it is appropriate to limit the inspection to one steam generator on a rotating schedule. Economic savings as well as reductions in personnel exposure and outage duration can be realized.

<u>Steam Generator Tube Sample Selection and Inspection (TS 4.2.b.2)</u>

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.

<u>Inspection Frequencies (TS 4.2.b.3)</u>

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.

Plugging or Repair Limit Criteria (TS 4.2.b.4)

Steam generator tubes found with less than the minimum wall thickness criteria determined by analysis, as described in WCAP-7832(2)(3), must either be repaired to be kept in service or removed from service by plugging.

Steam generator tube plugging is a common method of preventing primary-to-secondary steam generator tube leakage and has been utilized since the inception of PWR nuclear reactor plants. This method is relatively uncomplicated from a structural/mechanical standpoint as flow is cut off from the affected tube by plugging it in the hot and cold leg faces of the tubesheet.

To determine the basis for the sleeve plugging limit, the minimum sleeve wall thickness was calculated in accordance with Draft Regulatory Guide 1.121 (August 1976). In addition, a combined allowance of 20% of wall thickness is assumed for eddy current testing inaccuracies and continued operational degradation per Draft Regulatory Guide I.121 (August 1976).

Repair by sleeving, or other methods, has been recognized as a viable alternative for isolating unacceptable tube degradation and preventing tube leakage. Sleeving isolates unacceptable degradation and extends the service life of the tube, and the steam generator. Tube repair, by sleeving in accordance with WCAP-11643(4) has been evaluated and analyzed as acceptable. This WCAP establishes hydraulic equivalency ratios for the application of normal operating, upset, and accident condition bounding analyses. Design, installation, testing, and inspection of steam generator tube sleeves requires substantially more engineering than plugging, as the tube remains in service. Because of this, the NRC has defined steam generator tube repair to be an Unreviewed Safety Question as described in 10 CFR 50.59(a)(2). As such, other tube repair methods will be submitted under 10 CFR 50.90; and in accordance with 10 CFR 50.91 and 50.92, the Commission will review the method, issue a significant hazards determination, and amend the facility license accordingly. A 90-day time frame for NRC review and approval is expected.

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(2)WCAP-7832, "Evaluation of Steam Generator Tube, Tube Sheet, and Divider Plate Under Combined LOCA Plus SSE Conditions."

(3)E. W. James, WPSC, to A. Schwencer, NRC, dated September 6, 1977.

(4)WCAP-11643, Kewaunee Steam Generator Sleeving Report, Revision 1, November 1988 (Proprietary).

Tube Support Plate Alternate Plugging Criteria (TS 4.2.b.5)

Implementation of the tube support plate alternate plugging criteria for the Kewaunee steam generators is supplemented by a 100% bobbin coil inspection requirement, a reduced operating leakage requirement, inspection guidelines to provide consistency in the voltage normalization, and the requirement to perform a rotating pancake coil inspection of bobbin coil indications > 1.5 volts at the tube support plate intersections in order to characterize the principal degradation mechanism as ODSCC. For all cases, the tubes dispositioned for continued service will be based upon standard bobbin probe signal amplitude. The plant specific guidelines will be amended as appropriate to accommodate the information necessary to evaluate tube support plate signals with respect to the voltage parameters. Pending incorporation of the voltage verification requirement in ASME standard verifications, an ASME standard calibration against a laboratory standard will be used during inspections for consistent voltage normalization.

In addition, a potential steam line break leakage will be calculated for tubes left in service to demonstrate that the cumulative leakage is < 260 gpm for each steam generator. The methodology for calculating expected leak rates from projected crack distribution may be a probabilistic method as discussed in WCAP-12985[5], or a bounding calculation as discussed in EPRI's draft report[6].

Reports (TS 4.2.b.6)

Category C-3 inspection results are considered abnormal degradation to a principal safety barrier and are therefore reportable under 10 CFR 50.72(b)(2)(i) and 10 CFR 50.73(a)(2)(ii).

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(5) WCAP-12985, "Kewaunee Steam Generator Tube Plugging Criteria for ODSCC at Tube Support Plates," November 1991.

(6) EPRI Draft Report "PWR Steam Generator Tube Repair Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates", July 1991.

ATTACHMENT 3

To

Proposed Amendment No. 105

Letter from C.R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Technical Justification