April 17, 1995

Mr. M. L. Marchi
Manager - Nuclear Business Group
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, WI 54307-9002

SUBJECT: AMENDMENT NO. 118 TO FACILITY OPERATING LICENSE NO. DPR-43 - KEWAUNEE NUCLEAR POWER PLANT (TAC NO. M90879)

Dear Mr. Marchi:

The Commission has issued the enclosed Amendment No. 118 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant (KNPP). This amendment revises the Technical Specifications (TS) in response to your application dated November 8, 1994, as supplemented on January 9, February 14, March 8, and April 3, 1995.

The amendment revises the KNPP TS 3.1.d, "Leakage of Reactor Coolant," TS 4.2.b, "Steam Generator Tubes," and TS 3.4.a, "Steam Generators," to allow application of a voltage-based repair limit for the steam generator (SG) tube support plate (TSP) intersections experiencing outside diameter stress corrosion cracking (ODSCC). The amendment also reduces the allowed primary-to-secondary operational leakage from any one steam generator from 500 gallons per day (gpd) to 150 gpd. These changes to the tube repair criteria are applicable for the 1995 to 1996 operating cycle (Cycle 21) only.

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,

Original signed by:

Richard J. Laufer, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures:
1. Amendment No. 118 to License No. DPR-43
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 17, 1995

Mr. M. L. Marchi
Manager - Nuclear Business Group
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, WI 54307-9002

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Docket No. 50-305

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2. Safety Evaluation

cc w/encls: See next page

cc:

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Mr. Harold Reckelberg, Chairman
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 118
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 November 8, 1994, as supplemented on January 9, February 14, March 8, and April 3, 1995, 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.118 , 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

Richard J. Laufer

Richard J. Laufer, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: April 17, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 118

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 ii	TS ii
TS 3.1-9	TS 3.1-9
TS B3.1-10	TS B3.1-10
TS B3.1-11	TS B3.1-11
TS B3.1-12	TS B3.1-12
TS 3.4-1	TS 3.4-1
TS 3.4-3	TS B3.4-1
TS 3.4-4	TS B3.4-2
TS 3.4-5	---------
TS 4.2-3	TS 4.2-3
TS 4.2-4	TS 4.2-4
TS 4.2-5	TS 4.2-5
TS 4.2-6	TS 4.2-6
--------	TS 4.2-7
--------	TS 4.2-8
TS B4.2-4	TS B4.2-4

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 when the tube support plate, voltage-based repair criteria is applied. Primary to secondary leakage is limited to 150 gallons per day through any one steam generator when the tube support plate voltage-based repair criteria is applied. 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.

Leakage of Reactor Coolant (TS 3.1.d)[18]

TS (TS 3.1.d.1)

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 $91/\bar{E}\,\mu$ Ci/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 $X/Q = 2.0 \times 10^{-6}$ 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. Operators would then investigate the source of the leak and take actions necessary to isolate it. Should the leak result in a continuous discharge to the atmosphere via the component cooling surge tank and waste holdup tank, 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 (12) 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.

TS 3.1.d.2

The 150 gpd leakage limit through any one steam generator is specified to ensure tube integrity is maintained in the event of a main steam line break or under loss-of-coolant accident conditions. This reduced operational leakage rate is applicable in conjunction with the tube support plate voltage-based plugging criteria as specified in TS 4.2.b.5.

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[18] USAR Sections 6.5, 11.2.3, 14.2.4

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 are 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.[19]

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[20]. 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.

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.[21][22]

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

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[19] USAR Section 4.2

[20] USAR Section 9.2

[21] USAR Table 3.2-1

[22] USAR Figure 3.2-8

3.4 STEAM AND POWER CONVERSION SYSTEM

APPLICABILITY

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

OBJECTIVE

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

SPECIFICATION

a. Steam Generators

1. The reactor shall not be heated > 350°F unless the following conditions are satisfied.

   A. Two steam generators are OPERABLE.

      1. System piping and valves directly associated with providing auxiliary feedwater flow to the steam generators are OPERABLE.

      2. Five main steam safety valves per OPERABLE steam generator are OPERABLE, except during required surveillance tests or during in-service testing of these valves and steam generators in accordance with 10 CFR 50.55a, provided that at least two main steam safety valves associated with the steam generator under test are OPERABLE.

   B. A minimum of 39,000 gallons of water is available in the condensate storage tanks and the Service Water System is capable of delivering an unlimited supply from Lake Michigan.

   C. The DOSE EQUIVALENT I-131 on the secondary side of the steam generators does not exceed 0.1 μCi/cc.

2. If, when the reactor is > 350°F, any one of the conditions of TS 3.4.a.1 cannot be met within 48 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 and maintain the Reactor Coolant System < 350°F within an additional 12 hours

BASIS

Steam Generators (TS 3.4.a)

Two steam generators are required to be OPERABLE when the average reactor coolant temperature is > 350°F to ensure that sufficient heat removal capability exists for power operation and decay heat removal. Although one steam generator would provide sufficient decay heat removal capability, two steam generators are required in order to provide the necessary redundancy to meet the single failure criterion. An OPERABLE steam generator is defined by TS 3.4.a.

The ten main steam safety valves (five per steam generator) have a total combined rated capability of 7,660,380 lbs./hr at 1181 lbs. pressure. The maximum full-power steam flow at 1721 MWTH is 7,449,000 lbs./hr; therefore, the main steam safety valves will be able to relieve the total maximum steam flow if necessary. The requirement that five main steam safety valves per OPERABLE steam generator are available will assure sufficient steam relief capability.

Testing of the main steam system while the plant is in HOT SHUTDOWN conditions is permitted provided that at least two main steam safety valves associated with the steam generator under test are available to provide sufficient relief capacity to protect the system during the test.

The specified minimum water supply in the condensate storage tanks is sufficient for 4 hours of decay heat removal. The 4 hours are based on the Kewaunee site specific station blackout (loss of all AC power) coping duration requirement. When AC power is available, unlimited replenishment of the condensate storage supply is available from Lake Michigan through the Service Water System.

An evaluation was performed to determine the maximum permissible steam generator primary-to-secondary leak rate during a steam line break event. The evaluation considered both a preaccident and accident initiated iodine spike. The results of the evaluation show that the accident initiated spike yields the limiting leak rate. This evaluation was based on a 30 REM thyroid dose at the site boundary and initial primary and secondary coolant iodine activity levels of 1.0 μCi/gm and 0.1 μCi/gm DOSE EQUIVALENT I-131 respectively. A leak rate of 34.0 gpm was determined to be the upper limit for allowable primary-to-secondary leakage in the steam generator faulted loop. The steam generator in the intact loop was assumed to leak at a rate of 0.1 gpm, the standard operating leakage limit applied for the tube support plate voltage-based plugging criteria specified in TS 4.2.b.5.

Auxiliary Feedwater Pumps (TS 3.4.b)

In the unlikely event of complete loss of electrical power to the plant, continued capability of decay heat removal would be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary feedwater pumps, and by steam discharge to the atmosphere through the main steam safety valves. Each motor-driven pump is normally aligned to both steam generators; the discharge of the turbine-driven pump, which starts automatically, is aligned to backup both motor-driven pumps. Any single auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the reactor.

It is acceptable to exceed 350°F with an inoperable turbine-driven auxiliary feedwater pump. However, operability of the pump must be demonstrated within 72 hours after exceeding 350°F or a plant shutdown must be initiated.

With no auxiliary feedwater pumps OPERABLE, action shall be taken to restore a pump as soon as possible. The action with three pumps inoperable is to maintain the plant in an operating condition in which the auxiliary feedwater system is not needed for heat removal. When one pump is restored, then the LIMITING CONDITIONS FOR OPERATION specified in TS 3.4.b.2 are applied. Should the plant shutdown be initiated with no auxiliary feedwater pumps available, there would be no feedwater to the steam generator to cool the plant to 350°F when the Residual Heat Removal System could be placed in operation.

Turbine Overspeed Protection System (TS 3.4.c)

Turbine overspeed protection is provided to limit the possibility of turbine missiles. Overspeed protection is provided by three independent systems based on diverse operating principles. The three systems are the electro-hydraulic (E-H) system, the mechanical trip system, and the redundant overspeed trip system (ROST). The E-H and mechanical systems are single channel and operate on a one-out-of-one to trip logic; the ROST system is a three channel system, requiring two-out-of-three channels to trip.

REFERENCES

USAR Section 10
USAR Section 14.1

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 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 below and in TS 4.2.b.2.b.

   Tubes left in service as a result of application of the tube support plate plugging 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.

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.

   Implementation of the steam generator tube support plate voltage-based plugging criteria requires a 100% bobbin coil inspection for hot leg and cold leg tube support plate intersections down to the lowest cold leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of tubes inspected over their full length.

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 pre-service 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.1.C 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.

4. Plugging 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 intersections for which voltage-based plugging criteria are applied.(2)

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. Tube repair shall be in accordance with the methods described in WCAP-11643, "Kewaunee Steam Generator Sleeving Report (Mechanical Sleeves)" or CEN-413-P, "Kewaunee Steam Generator Tube Repair Using Leak Tight Sleeves."

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

(2)The tube support plate voltage-based repair criteria is applicable for the 1995 to 1996 operating cycle only.

c. Any Combustion Engineering leak tight sleeve which, upon inspection, exhibits wall degradation of 40% or more shall be plugged prior to returning the steam generator to service. This plugging limit applies to the sleeve up to and including the weld region.

5. Tube Support Plate Voltage-Based Plugging Criteria[3]

The following criteria are used for the disposition of a steam generator tube for continued service that is experiencing outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersection, the repair limit is based on maintaining steam generator tube serviceability as described below:

a. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage ≤ 2.0 volts will be allowed to remain in service.

b. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage > 2.0 volts will be repaired or plugged except as noted in TS 4.2.b.5.c below.

c. Indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage > 2.0 volts but ≤ 5.6 volts may remain in service if a rotating pancake coil inspection does not detect degradation. Indications of outside diameter stress corrosion cracking degradation with a bobbin voltage > 5.6 volts will be plugged or repaired.

d. If, as a result of leakage due to a mechanism other than ODSCC at the tube support plate intersection or some other cause, an unscheduled mid-cycle inspection is performed, the following repair criteria apply instead of TS 4.2.b.5.c. If bobbin voltage is within expected limits, the indication can remain in service. The expected bobbin voltage limits are determined from the following equation:

$$V < \frac{\frac{\Delta t}{CL}(V_{SL} - V_{BOC}) + V_{BOC}}{1 + (.2)\left(\frac{\Delta t}{CL}\right)}$$

[3]The tube support plate voltage-based repair criteria is applicable for the 1995 to 1996 operating cycle only.

Where:

V = measured voltage
V_{BOC} = voltage at BOC
Δt = time period of operation to unscheduled outage
CL = cycle length (full operating cycle length where operating cycle is the time between two scheduled steam generator inspections)
V_{SL} = 9.6 volt for 7/8 inch tubes

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 Specification 4.2.b.6.a, using the Licensee Event Report System to satisfy the intent of 10 CFR 50.73(a)(2)(ii).

d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the actual measured end-of-cycle voltage distribution would have exceeded the leak limit (for the postulated main steam line break utilizing licensing basis assumptions) during the previous operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines the tube support plate.

4. If the calculated conditional burst probability exceeds the threshold value, notify the NRC and provide an assessment of the safety significance of the occurrence.

There are three types of Combustion Engineering leak tight sleeves. The first type, the straight tubesheet sleeve, spans the degraded area of the parent tube in the tubesheet crevice region. The sleeve is welded to the parent tube near each end. The second type of sleeve is the peripheral tubesheet sleeve. The sleeve is initially curved as part of the manufacturing process and straightened as part of the installation process. The third type of sleeve, the tube support plate sleeve, spans the degraded area of the tube support plate and is installed up to the sixth support plate. This sleeve is welded to the parent tube near each end of the sleeve.

The hydraulic equivalency ratios for the application of normal operating, upset, and accident condition bounding analyses have been evaluated. 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 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.

<u>Technical Specification 4.2.b.5</u>[5]

The repair limit of tubes with degradation attributable to outside diameter stress corrosion cracking contained within the thickness of the tube support plates is conservatively based on the analysis documented in WCAP-12985, "Kewaunee Steam Generator Tube Plugging Criteria for ODSCC at Tube Support Plates" and EPRI Draft Report TR-100407, Rev.1, "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates." Application of these criteria is based on limiting primary-to-secondary leakage during a steam line break to ensure the applicable 10 CFR Part 100 limits are not exceeded.

<u>Technical Specification 4.2.b.6</u>

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]The tube support plate voltage-based repair criteria is applicable for the 1995 to 1996 operating cycle only.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 118 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 November 8, 1994, as supplemented on January 9, February 14, March 8, and April 3, 1995, Wisconsin Public Service Corporation (WPSC), the licensee, requested a revision to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS). The proposed amendment would revise the KNPP TS 3.1.d, "Leakage of Reactor Coolant," TS 4.2.b, "Steam Generator Tubes," and TS 3.4.a, "Steam Generators," to permit the use of a voltage-based steam generator tube repair criteria for defects confined to within the thickness of the tube support plate. The amendment would also reduce the allowed primary-to-secondary operational leakage from any one steam generator from 500 gallons per day (gpd) to 150 gpd. All of the proposed changes to the tube repair criteria would be applicable for the 1995 to 1996 operating cycle (Cycle 21) only.

The proposed voltage-based tube repair criteria pertain specifically to outside diameter stress corrosion cracking (ODSCC) flaws. The proposed criteria would: (1) permit flaws confined to within the thickness of the tube support plate with bobbin voltages less than or equal to 2.0 volt to remain in service; (2) permit flaws confined to within the thickness of the tube support plate with bobbin voltages greater than 2.0 volt but less than or equal to 5.6 volts to remain in service if a rotating pancake coil (RPC) probe does not detect degradation; and (3) require flaw indications confined to within the thickness of the tube support plate with bobbin voltages greater than 5.6 volts to be plugged or repaired.

Additional clarifying information with respect to implementation of the voltage-based tube repair criteria was provided in the licensee's letters dated January 9, February 14, March 8, and April 3, 1995.

2.0 BACKGROUND

The NRC staff is currently developing a generic interim position on voltage-based limits for ODSCC confined to within the thickness of the tube support plates. The staff has published several conclusions regarding voltage-based repair criteria in draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes," and in a draft generic letter titled "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes." The latter document was published for public comment in the Federal Register on August 12, 1994. However, the staff is continuing to evaluate an acceptable generic position which will take into consideration public comments on the draft generic letter cited above, domestic operating experience under the voltage-based repair criteria, and additional data which have been made available from European nuclear power plants. The staff currently plans to document its final position on this matter in a generic letter. Pending completion and issuance of the staff's final generic position on the voltage-based tube repair criteria, the staff is continuing to evaluate voltage-based repair criteria proposals on a case-specific basis, as necessary, to ensure that there is adequate assurance of public health and safety. Furthermore, these case-specific evaluations limit the applicability of the voltage-based repair criteria to one cycle of operation.

In a letter dated November 8, 1994, the licensee requested an amendment to modify the technical specifications to allow the use of a voltage-based steam generator tube repair criteria. Based on subsequent discussions between the licensee and the NRC staff, the licensee provided a revised amendment request by letter dated March 8, 1995, which modified the request to apply only to Cycle 21 and provided clarifying information. Additional clarifying information was also provided in letters dated January 9, February 14, and April 3, 1995.

The tube repair limits proposed by the licensee include a lower voltage repair limit of 2.0 volts for axially oriented ODSCC flaws confined to within the thickness of the tube support plates in lieu of the present criteria which is a depth-based limit of 40% or 50% depending on the degradation mechanism. In addition, the repair limits allow bobbin indications between 2.0 and 5.6 volts (the upper voltage repair limit) to remain in service provided inspection of these indications with a RPC probe does not confirm the degradation to be present.

The licensee's proposal is similar to that reviewed and approved for several other plants and has been reviewed on a case-specific basis. The tube structural limit is based on maintaining a margin of safety of 1.43 against tube failure under postulated accident conditions and maintaining a margin of safety of 3 against burst during normal operation. The margin of safety of 3 against burst during normal operation is inherently satisfied since the structural constraint provided by the tube support plates, which surround the degradation to which the voltage-based repair criteria applies, ensures these tubes will maintain this margin of safety at these locations. To complement these deterministic criteria, the conditional probability of burst under accident conditions and the primary-to-secondary leakage from the steam generator tubes during a postulated main steam line break (MSLB) are also calculated.

3.0 PROPOSED INTERIM TUBE REPAIR CRITERIA

Kewaunee Technical Specifications 3.1.d, 4.2.b.2, 4.2.b.4, 4.2.b.5, and 4.2.b.6 and Bases 3.1.d, 4.2.b.5, and 4.2.b.6, would be revised by this proposed amendment to specify the tube repair and leakage criteria for ODSCC confined to within the thickness of the tube support plate. The proposed changes to the tube repair and leakage criteria in the technical specifications specify, in part:

a. Implementation of the steam generator tube support plate voltage-based plugging criteria requires a 100% bobbin coil probe inspection for all hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known ODSCC indications. The determination of the tube support plate intersections having ODSCC indications shall be based on the performance of at least 20% random sampling of tubes inspected over their full length.

b. Degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage ≤ 2.0 volts will be allowed to remain in service.

c. Degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage > 2.0 volts will be repaired or plugged except as noted in Item (d) below.

d. Indications of potential degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage > 2.0 volts but ≤ 5.6 volts may remain in service if a RPC inspection does not detect degradation. Indications of ODSCC degradation with a bobbin voltage > 5.6 volts will be plugged or repaired.

e. If, as a result of leakage due to a mechanism other than ODSCC at the tube support plate intersections or some other cause, an unscheduled mid-cycle inspection is performed, the following repair criteria apply instead of Item (d) above. If the bobbin voltage is within expected limits, the indication can remain in service. The expected bobbin voltage limits are determined from the following equation:

$$V < \frac{\frac{\Delta t}{CL}\left(V_{SL} - V_{BOC}\right) + V_{BOC}}{1 + (.2)\left(\frac{\Delta t}{CL}\right)}$$

where:

V = bobbin voltage
V_{BOC} = voltage at the beginning of cycle (BOC)
Δt = time period of operation to unscheduled outage
CL = cycle length (full operating cycle length where the operating cycle is the time between two scheduled steam generator inspections)
V_{SL} = 9.6 volts for 7/8-inch tubes

f. For implementation of the voltage-based repair criteria to tube support plate intersections, notification of the NRC staff prior to returning the steam generators to service is required should any of the following conditions arise:

   (1) If the estimated leakage based on the actual measured end-of-cycle (EOC) voltage distribution would have exceeded the leak limit (for the postulated MSLB using licensing basis assumptions) during the previous operating cycle.

   (2) If circumferential crack-like indications are detected at the tube support plate intersections.

   (3) If indications are identified that extend beyond the confines of the tube support plate.

   (4) If the calculated conditional burst probability exceeds the threshold value. Additionally, an assessment of the safety significance of this occurrence should be provided.

g. Reactor coolant-to-secondary leakage through the steam generator tubes shall be limited, in part, to 150 gpd through any one steam generator when the tube support plate voltage-based repair criteria is applied.

In addition to the above proposed technical specification changes, the licensee also made the following commitments for implementing the voltage-based repair criteria:

1. All bobbin indications with voltages greater than 1.5 volts will be inspected with a RPC probe. RPC probe inspections assist in identifying axial ODSCC as the dominant mechanism for indications at the tube support plates.

2. Tubes with bobbin dent voltages exceeding 5.0 volts, large mixed residual, or indications of copper deposits will be inspected with a RPC and any RPC flaw indications detected at these intersections will be dispositioned in accordance with the depth-based repair criteria.

3. Tubes with known leaks will be repaired prior to returning the steam generators to service.

4. Steam generator tube integrity data (i.e., voltage distributions and leak/burst evaluations) will be provided to the NRC within 90 days following restart.

5. A 0.720-inch diameter bobbin coil probe will be used during the steam generator inspections at intersections where the voltage-based repair criteria will be applied.

6. The NRC will be notified prior to plant restart if any primary water stress corrosion cracking (PWSCC) indications are detected within the tube support plate intersections during the steam generator inspection. Additionally, the eddy current analysts will be briefed on the potential that PWSCC can occur at the tube support plate locations.

7. The conditional probability of burst and the primary-to-secondary leakage calculation will be performed in accordance with the guidance provided in the draft generic letter using the methodology described in WCAP-14277.

8. The conditional probability of burst calculation will be compared against a threshold value of $1x10^{-2}$.

In general, the licensee intends to follow the guidance of the draft generic letter with the following exceptions: (1) calibration of the bobbin coil probe on the 4-20% through-wall holes rather than the 4-100% through-wall holes; (2) implementation of the probe wear standard; (3) limiting new probe variability; (4) removing specimens for destructive examination; and (5) the application of data exclusion criteria. These exceptions are discussed below.

4.0 EVALUATION

4.1 Inspection Issues

In support of the proposed voltage-based repair limits, the licensee proposes to utilize the eddy current test guidelines included as Appendix A to WCAP-12985, Revision 2, dated March 1993, and as later supplemented. The inspection criteria are intended to ensure the inspection scope, data acquisition, and data analysis are performed in a manner consistent with the methodology utilized to develop the voltage limits. The proposed guidelines define, in part, the bobbin specifications, calibration requirements, specific acquisition and analyses criteria, and flaw recording guidelines to be used for the inspection of the steam generators.

The inspections to be performed as part of the voltage-based repair criteria include both bobbin coil and rotating pancake coil (RPC) examinations. Bobbin coil examinations will be performed for 100% of the hot-leg tube support plate intersections and cold-leg intersections down to the lowest cold-leg tube support plate with known ODSCC. The determination of the tube support plate having ODSCC indications will be based on a minimum 20% random sampling of the tubes over their full length. The bobbin coil examinations for intersections at which the voltage-based repair criteria will be applied will be performed with a 0.720-inch bobbin coil probe. RPC examinations will be performed to permit additional characterization of the flaws found with the bobbin coil probe and to inspect intersections with significant bobbin interference signals (due to copper deposits, dents, large mix residuals) which may impair the ability of the bobbin coil probe to detect flaws or which may unduly influence the bobbin voltage measurement.

With respect to flaw characterization, a key purpose of the RPC inspections is to ensure the absence of detectable crack-like circumferential indications and detectable indications extending outside the thickness of the tube support plate. The voltage-based repair criteria are not applicable to intersections exhibiting such indications (i.e., circumferential indications and indications extending outside the tube support plates), and special reporting requirements pertaining to the finding of such indications have been proposed if these types of indications are detected. RPC examinations will be performed (1) at all intersections with bobbin coil indications exceeding 1.5 volts, (2) at all intersections where the dent signal is greater than 5.0 volts as measured with the bobbin coil probe, (3) at intersections where the mixed residual could cause a 1.0 volt bobbin signal to be missed or misread (i.e., masked), and (4) at all intersections where copper deposits influence the bobbin coil signal. Any flaw-like indications found at intersections with dent signals greater than 5.0 volts, with large mixed residuals, or where copper deposits influence the bobbin coil signal will be dispositioned in accordance with the depth-based tube repair criteria.

As previously mentioned, tube support plate locations with bobbin dent voltages above 5.0 volts, as measured by the bobbin coil probe, will be inspected with an RPC probe. Inspections of dented intersections are performed, in part, as a result of (1) the possible masking effect the dent may have on the detection of flaw indications, (2) the possible development of primary water stress corrosion cracking (PWSCC) flaws at these locations, and (3) the possible development of circumferential cracks at these locations. With respect to masking flaw indications, it is anticipated that flaw signals on the order of 1.0 volt would have phase angles that fall within the flaw reporting range even if the bobbin dent voltage was as high as 5.0 volts based on a vectorial combination of the eddy current signals attributed to the flaw and to the dent. As a result, RPC inspecting all intersections with bobbin dent voltages in excess of 5.0 volts provides reasonable assurance that any structurally significant ODSCC indications will be detected and repaired. With respect to the occurrence of circumferential cracking at the support plate elevations, the RPC sampling plan provides assurance that if a significant amount of circumferential cracking is occurring at the tube support plate elevations it will be detected.

With respect to the occurrence of PWSCC at dented tube support plate intersections, the potential exists for axial PWSCC to occur at intersections where the bobbin dent voltage is less than 5.0 volts. Most frequently these types of indications (i.e., indications representative of axially oriented PWSCC) have been found at tube support plates with significant denting, have been known to occur at 180° spacing as two axial indications due to the stresses in the tube, and have been known to occur within the tube support plate but occasionally extending outside the tube support plate. Axial PWSCC is not presently analyzed as part of the voltage-based repair criteria. As a result of this and the potential for PWSCC to occur at dented intersections less than 5.0 volts, the licensee has proposed to (1) RPC inspect all bobbin indications which are greater than 1.5 volt at dented intersections (2) RPC inspect all intersections where the bobbin dent voltage is greater than 5.0 volts regardless of whether a bobbin indication is detected, and (3) notify the NRC prior to plant restart if any PWSCC indications are detected at the support plate elevations. In addition, the licensee will brief the eddy current analysts on the potential for PWSCC at tube support plate locations and the analysts will be instructed to report occurrences of axial PWSCC. The staff finds this sampling plan adequate to detect the onset of axial PWSCC at support plate locations. The staff also notes that frequently axial PWSCC extends outside the tube support plate intersection, making it more likely to be detectable with the bobbin coil. This provides added confidence that if extensive axial PWSCC is present, it will be detected. The staff notes that if PWSCC is detected at support plate elevations, an evaluation to ensure the voltage-based repair criteria is only applied to ODSCC indications will need to be performed and reviewed by the staff.

With respect to data acquisition and analysis, the licensee's eddy current guidelines either contain requirements or guidance pertaining to (1) recording all indications regardless of voltage amplitude, (2) controlling probe wear by the use of a probe wear standard, (3) calibrating the bobbin coil probes, and (4) using a transfer standard to ensure consistency between the voltages measured in the field and the voltages measured in the laboratory as part of the development of the voltage-based approach.

The staff notes that there are several outstanding technical issues with respect to the inspection guidelines, as documented in previously issued NRC documents (e.g., in draft NUREG-1477 and in the draft generic letter cited above) which will be resolved prior to issuing the final generic letter on voltage-based limits for ODSCC confined to within the thickness of the tube support plate. These outstanding issues include, in part, (1) limits on new probe variability, (2) the need to reinspect all tubes since the last successful probe wear check, (3) the need to calibrate the bobbin coil on the 4-100% holes versus the 4-20% holes, and (4) the capabilities/limitations of the 1-coil, 2-coil, and 3-coil RPC probes. However, the staff concludes that the inspection guidelines submitted by the licensee are acceptable since the proposed repair criteria is limited to one cycle, and the calibration, recording, and analysis requirements are consistent with the methodology used in the development of the tube repair criteria described in the draft generic letter.

4.2 Tube Integrity Issues

The thin-walled tubing of the steam generator constitutes more than half of the reactor coolant pressure boundary (RCPB), and maintenance of the structural and leakage integrity of this boundary is a requirement under Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50), Appendix A. Specific requirements governing the maintenance of steam generator tube integrity are contained in the plant technical specifications and Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code). These include requirements for periodic inservice inspection of the tubing, flaw acceptance criteria (i.e., repair limits for plugging or sleeving), and primary-to-secondary leakage limits. These requirements, coupled with the broad scope of plant operational and maintenance programs, have formed the basis for assuring adequate steam generator tube integrity.

Flaw acceptance criteria, termed plugging/repair limits, are specified in the plant technical specifications. The purpose of the technical specification repair limits is to ensure that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions, consistent with General Design Criteria 14, 15, 30, 31 and 32 of 10 CFR Part 50, Appendix A. Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the steam generator tubing. Leakage integrity refers to limiting primary-to-secondary leakage to within acceptable limits.

The traditional strategy for accomplishing the objectives of the General Design Criteria related to steam generator tube integrity has been to establish a minimum wall thickness requirement in accordance with the structural criteria of Regulatory Guide 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes." Allowances for eddy current measurement error and flaw growth between inspections have been added to the minimum wall thickness requirements, consistent with Regulatory Guide 1.121, to arrive at a depth-based repair limit. Development of the minimum wall thickness requirements to satisfy Regulatory Guide 1.121 was governed by analyses for uniform thinning of the tube wall in the axial and circumferential directions. The assumption of uniform thinning conservatively bounds the degrading effects of all flaw types currently occurring in the field and is the basis of the standard 40% depth-based repair limit incorporated into the technical specifications. However, the 40% repair limit is conservative for highly localized flaws such as pits and short cracks. In particular, the 40% depth-based repair limit is conservative for ODSCC that occurs at the tube support plate intersections.

Enforcement of a minimum wall thickness requirement for the steam generator tubes would implicitly serve to ensure leakage integrity during normal operation and postulated accidents, as well as structural integrity. It has been recognized, however, that defects, especially cracks, may occasionally grow entirely through-wall and develop small leaks. For this reason, limits on the allowable primary-to-secondary leakage have been established in a plant's technical specifications to ensure timely plant shutdown before adequate structural and leakage integrity of an affected steam generator tube is impaired.

The proposed voltage-based tube repair limits consist of voltage amplitude criteria rather than the traditional depth-based criteria. Thus, the repair criteria represents a departure from the past practice of explicitly enforcing a minimum wall thickness requirement.

The industry-wide database from examination of steam generator tubes removed from a number of steam generators in operating nuclear power plants shows that for bobbin indications exceeding 2.0 volts (i.e., the lower voltage repair limit), maximum crack depths range between 50% and 100% through-wall. The likelihood of through-wall or near through-wall crack penetrations appears to increase with increasing voltage amplitude. For indications at or near 5.6 volts (i.e., the upper voltage repair limit), the maximum crack depths have been found to generally range between 90% and 100% through-wall. Many of the tubes which will be allowed to remain in service under the proposed voltage-based repair criteria may have or develop through-wall or near through-wall crack penetrations during the upcoming cycle, thus creating the potential for leakage during normal operation and postulated MSLB accidents.

The staff's evaluation of the proposed repair criteria from a structural and leakage integrity standpoint is provided in Sections 4.3 and 4.4 of this evaluation.

Although the voltage-based repair limits ensure adequate structural and leakage integrity, the NRC staff recognizes that overall margins have been reduced when compared to the margins associated with the existing 40% depth-based repair limit. Because of the increased likelihood of through-wall cracks developing in service, the staff has included provisions for augmented steam generator inspections, as discussed in the previous section, and more restrictive operational tube leakage limits, as discussed below.

4.3 Structural Integrity

4.3.1 Deterministic Structural Integrity Assessment

The licensee has proposed a burst pressure/bobbin voltage correlation to demonstrate that bobbin indications satisfying the 2.0 volt lower voltage repair limit would retain adequate structural margins, consistent with the criteria of Regulatory Guide 1.121. The correlation was developed from both pulled steam generator tube data from other plants (using pre-pull bobbin voltages) and laboratory tube specimens containing ODSCC flaws. The bobbin voltage data used to construct the burst pressure/bobbin voltage correlation were normalized and are consistent with the calibration standard voltage set-ups and voltage measurement procedures to be used by the licensee during the steam generator inspections.

To confirm the nature of the degradation occurring at the tube support plate elevations, the licensee pulled three tubes with five tube support plate intersections from the steam generators during an outage in the Spring of 1993. Tube pulls confirm that the nature of the degradation being observed at the tube support plate elevations is predominantly axially oriented ODSCC and also provide data for assessing the reliability of the inspection methods and for supplementing existing databases (e.g., burst pressure, probability of leakage, and leak rate). Destructive examination of these tube support plate intersections was performed. The examinations performed confirmed that the dominant degradation mechanism for the indications at the support plate elevations was axially oriented ODSCC and that the voltage-based tube repair criteria for indications at the tube support plates was applicable at Kewaunee.

The draft generic letter on voltage-based repair criteria provides guidance on performing tube pulls for initial implementation of the repair criteria. In summary, the draft generic letter states that at least six tube support plate intersections should be obtained either during the outage in which the voltage-based repair criteria is implemented or during the inspection outage preceding initial application of the voltage-based repair criteria. To follow the draft generic letter guidance on tube pulls, the licensee would need to pull 6 intersections from their steam generators during this outage since their last tube pulls were two outages ago. The current guidance in the draft generic letter on the issue of tube pulls gives no consideration to the length of the operating interval between inspections and can result in plants with short operating intervals removing more tubes (in the long run) than a plant with longer operating intervals. As a result of this and other public comments received on this issue, the staff has been evaluating alternative options to the tube pull guidance in the draft generic letter. The latest guidance was presented to the industry during a public meeting on January 18, 1995. The licensee believes their tube pulls met the intent of this guidance as discussed in a letter from the licensee dated February 14, 1995 and, as a result, the licensee does not intend to pull tubes during the upcoming outage. Pending finalization of the generic letter position on tube pulls, the staff has concluded that the licensee need not remove tubes during the upcoming outage to meet the guidance in the draft generic letter.

The voltage-based tube repair criteria previously approved by the staff for other plants have been set deterministically to ensure that indications accepted for continued service with this repair criteria will retain adequate structural integrity during the full range of normal, transient, and postulated accident conditions. The repair criteria includes allowances for eddy current test uncertainty and flaw growth projected to occur during the next operating cycle. Because the voltage-based repair criteria addresses tubes affected with ODSCC confined to within the thickness of the tube support plates during normal operation, the staff has concluded that the structural constraint provided by the tube support plates ensures that all tubes to which the voltage-based criteria applies will retain a margin of 3 with respect to burst under normal operating conditions, consistent with the criteria of Regulatory Guide 1.121. For a postulated MSLB accident, however, the tube support plates may displace axially during blowdown such that the ODSCC affected portion of the tubing may no longer be fully constrained by the tube support plates. Accordingly, it is appropriate to consider the ODSCC affected regions of the tubes as free standing tubes for the purpose of assessing burst integrity under postulated MSLB conditions.

The allowable end-of-cycle (EOC) voltage which ensures a margin of 1.43 with respect to burst under postulated MSLB conditions (i.e., 3660 psi), in accordance with Regulatory Guide 1.121, is based on the lower 95% prediction interval of the burst pressure/bobbin voltage correlation, adjusted for lower bound material properties evaluated at the 95/95 confidence level. This voltage limit is approximately 9 volts for the 7/8-inch diameter tubing used in the Kewaunee steam generators. The difference between the 9 volt allowable EOC voltage and the 2.0 volt repair criterion represents an allowance of approximately 7 volts for voltage growth (i.e., ODSCC flaw growth) during the forthcoming fuel cycle (i.e., Cycle 21) and for eddy current voltage measurement variability (i.e., the repeatability error) during the steam generator inspection.

To demonstrate the adequacy of the voltage-based repair criteria, the largest RPC confirmed indication which may be left in service (i.e., a 2.0 volt indication), was analyzed by the staff to determine if the indication would grow to the point that the structural voltage limit (i.e., approximately 9 volts) is exceeded. In this analysis, a 2.0 volt bobbin indication is assumed to grow at a rate equal to the maximum growth rate observed during the latest cycle for which data is available (i.e., 1.24 volts for Cycle 19 which was 0.89 effective full power years (EFPY) in duration) and it is assumed that the indication was undersized by 20% (i.e., the 95% cumulative probability of the non-destructive examination (NDE) uncertainty). The resultant EOC voltage is determined from this analysis to be 4.2 volts for the 1.3 EFPY planned for Cycle 21. This EOC voltage compares favorably to the structural voltage limit determined from the burst pressure versus bobbin voltage correlation.

The proposed 2.0 volt lower repair limit is applicable to all bobbin indications confirmed by RPC or which have not been RPC inspected. The licensee is also proposing a 5.6 volt upper voltage repair limit applicable to bobbin indications which have been RPC inspected but for which the RPC failed to confirm the bobbin indication. This 5.6 volt upper voltage repair limit can be derived from the information in EPRI Report TR-100407, Revision 1, "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates," dated August, 1993. The maximum voltage which ensures a margin of 1.43 with respect to burst under postulated MSLB conditions (i.e., 3660 psi) for tubes with lower bound material properties at a 95% prediction interval was 9.6 volts based on the data available at that time. A 5.6 volt upper voltage repair limit was calculated from the 9.6 volt structural limit by including an allowance for average growth rates of 50% of the BOC voltage amplitude and an allowance of 20% for eddy current voltage measurement variability (i.e., the 95% cumulative probability of the NDE uncertainty).

Since the issuance of EPRI Report TR-100407, Revision 1 in August 1993, additional data has been added to the burst pressure database used in the development of this upper voltage repair limit and several of the existing data points in the database have been updated as a result of additional analysis. However, taking this into consideration with the growth rates and the planned operating interval for Kewaunee, the staff has concluded that the 5.6 volt upper voltage repair limit is adequate for this cycle of operation. The new upper voltage repair limit was calculated to be approximately 5.6 volts for Kewaunee assuming an allowance of approximately 40% for flaw/voltage growth over the next operating cycle (i.e., Cycle 21) and an allowance of 20% for measurement variability. The voltage measurement variability estimate considers measurement variabilities stemming from bobbin coil probe wear and variability in the analysts' interpretation of the bobbin coil voltage. Potential flaw growth between inspections has been evaluated based on observed voltage amplitude changes during prior cycles at Kewaunee. Over the last few cycles (typically between 0.8 and 1.0 EFPYs), the average percent voltage growth at Kewaunee has been 18% (1991 to 1992), 5% (1992 to 1993), and 13% (1993 to 1994). The 40% average growth allowance used to support the approximately 5.6 volt upper voltage repair limit is intended to provide margins for variation in future growth rates at Kewaunee and for the increased length of the operating interval (i.e., 1.3 EFPY). As a result of the above analysis, the staff concludes that the 5.6 volt upper voltage repair limit is acceptable for Kewaunee.

The staff has evaluated the acceptability of the upper voltage repair limit for indications below this limit which may be left in service if detected by the bobbin coil probe but not confirmed to be flaw-like by the RPC probe. Short and/or relatively shallow cracks detected by the bobbin coil may sometimes not be detectable by the RPC probe, although the RPC probe is considered by the staff to be more sensitive to longer, deeper flaws which are of structural significance. Furthermore, the burst strength of steam generator tubing affected by predominantly axially oriented ODSCC at the support plate elevations is not a unique function of the bobbin voltage. Rather, for a given voltage, there is a statistical distribution of possible burst strengths, as indicated in the burst pressure/bobbin voltage correlation. The staff believes that the burst pressure for bobbin indications which were not confirmed to be flaw-like by the RPC probe will tend to be at the upper end of the burst pressure distribution (i.e., exhibit a higher burst pressure). That is, ODSCC which is not detectable by RPC is believed to be less likely to affect the tube structural and leakage integrity during the operating cycle than ODSCC which is detectable by both the bobbin coil and the RPC probe. In addition, the burst and leakage potential for bobbin indications accepted for continued service under the 5.6 volt criterion have been directly considered in the probability of burst and leakage assessments described below, with no credit given to the fact that RPC failed to confirm the indications. Based on these considerations, the staff finds the upper voltage repair limit of 5.6 volts for indications which may be left in service if detected by bobbin inspection but not confirmed by the RPC to be acceptable.

4.3.2 Probabilistic Structural Integrity Assessment

A probabilistic analysis for the potential for steam generator tube ruptures, given a MSLB, must also be performed. The need for this analysis, which supplements the deterministic analysis discussed above, is dictated by the following considerations:

1. The deterministic analysis does not consider the tail of the burst pressure distribution beyond the lower 95% prediction interval used to determine the maximum allowable EOC voltage. Given the large numbers of indications which could potentially be accepted for continued service with the 2.0 volt criterion, the probabilistic analysis ensures that the use of the 95% prediction interval value in lieu of the 99% or 99.9% values does not lead to a significant likelihood of steam generator tube rupture given a MSLB.

2. The deterministic assessment ignores the burst and leakage potential of bobbin indications between 2.0 volt and 5.6 volts for which the RPC probe failed to confirm the indication. The probabilistic assessment, however, considers the burst potential of these indications with no credit given for the lack of confirmation by the RPC probe of the presence of these indications.

3. The deterministic analysis does not account for bobbin indications missed by the data analysts. The staff concluded in draft NUREG-1477 and in the draft generic letter that the probabilistic assessment is required in order to address the burst potential of indications missed by the data analysts.

4. The deterministic analysis does not consider the cumulative effect of the entire distribution of indications accepted for continued service.

Employing the probabilistic analysis, however, ensures that all indications accepted for continued service are accounted for in determining the overall probability of burst given a MSLB.

5. The deterministic analysis does not consider the tails of the material properties distribution and the eddy current voltage variability distributions. The probabilistic analysis does include the entire distribution of material properties and voltage variability.

To perform the probabilistic analysis, the EOC distribution of indications must be determined. Consistent with the approach recommended in the draft generic letter on voltage-based repair criteria, the BOC distribution used in the determination of the EOC distribution involves adjusting the indications detected during the inspection by the probability of detection (POD), where the POD is assumed to have a constant value of 0.6, irrespective of voltage. The net effect of this assumption is that the distribution of detected bobbin indications is scaled up by a factor of 1/POD. After this POD scaling is made, indications removed from service by tube repair (i.e., plugging or sleeving) are subtracted from this distribution to yield the assumed BOC distribution. The EOC distribution is then determined by combining the voltage measurement uncertainty distribution, the voltage growth rate distribution, and the BOC voltage distribution using Monte Carlo techniques. For each of the resultant EOC voltages determined by the above analysis, the distribution of burst pressures as a function of bobbin voltage along with a distribution of material properties is sampled by Monte Carlo techniques to yield a distribution of burst pressures for the EOC voltage distribution. The conditional probability of burst, given a MSLB, can then be determined by dividing the number of times the Monte Carlo analysis yields a burst pressure below the MSLB differential pressure for the EOC voltage distribution by the total number of samples. A distribution of material tensile properties is sampled in the probabilistic analysis since the data points in the bobbin voltage/burst pressure correlation have been normalized to a flow stress of 75 ksi.

The POD scaling approach cited above is reasonably consistent with reported operating experience to-date with ODSCC in terms of accounting for the projected distribution of indications at EOC which were not previously detectable at BOC. However, operating experience to-date, for ODSCC confined to within the thickness of the tube support plate, is that maximum EOC bobbin voltages generally do not exceed 4 or 5 volts. Although there are known cases where indications on the order of 3 volts have not been detected, there is very little experience regarding the likelihood of not detecting bobbin indications between 3 and 10 volts. The industry believes that the numerical value of the POD is substantially higher than 0.6 for indications exceeding 1.0 volt, based, in part, on data collected from the Electric Power Research Institute (EPRI) performance demonstration program. However, pending further staff review, the staff believes a POD value of 0.6 is appropriate for this voltage-based repair criteria application.

The licensee will perform the probabilistic analysis discussed above which assumes the degradation is free span and ignores the potential constraining effects of the tube support plates. In addition, this analysis will be performed in a manner which considers the uncertainty in the parameters for the supporting correlations (e.g., burst pressure/bobbin voltage correlation). The results of the probabilistic analysis will be compared to a threshold value established by the staff. Consistent with the draft generic letter this threshold value is $1x10^{-2}$. This threshold value will provide assurance that the probability of burst is acceptable considering the assumptions of the calculation and the results of the staff's generic risk assessment for steam generators contained in NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." Failure to meet the threshold value indicates that ODSCC confined to within the thickness of the tube support plate could contribute a significant fraction to the overall conditional probability of tube rupture from all forms of degradation that was assumed and evaluated as acceptable in NUREG-0844. In addition, the threshold value provides an indication that one or more tubes may not maintain the Regulatory Guide 1.121 safety margins for the entire operating cycle. The licensee has stated that the results of the probability of burst analysis will be compared against a threshold value of $1x10^{-2}$. If this threshold value is exceeded, the NRC staff will be notified and an assessment of the safety significance of this occurrence will be provided to the NRC staff prior to returning the steam generators to service. The staff notes that all applicable data should be included in the burst pressure database when performing this calculation, except as discussed below.

4.3.3 Data Exclusion from the Burst Pressure Correlation

During the performance of the pulled tube examinations, malfunctions in the test equipment or improper specimen preparation can occasionally occur which could result in erroneous readings. Data such as this should not be included in a database since it could result in invalid results and/or conclusions. The staff, therefore, concluded in draft NUREG-1477 that eliminating data from the bobbin voltage/burst pressure database was appropriate provided that the data could be shown to be erroneous or the result of an invalid test. The staff provided additional guidance regarding the exclusion of data from the correlations used in the bobbin voltage/burst pressure database in a meeting with the industry on February 8, 1994. As a result of this guidance, the industry provided criteria for determining whether data may be removed from the burst pressure/bobbin voltage database. The specific criteria are presented in a letter referenced by the licensee which was submitted to the NRC by the Electric Power Research Institute (EPRI) on April 22, 1994.

The data points excluded from the burst pressure/bobbin voltage database as a result of applying these criteria are listed in Table E-1 of the subject document. The staff has concluded that excluding the data points listed in Table E-1 from the 7/8-inch diameter steam generator tubing burst pressure/bobbin voltage database is appropriate since it meets the exclusion criteria discussed by the staff at the February 8, 1994, industry meeting. Pending further evaluation of the generic criteria presented in Section E.2, the staff is continuing to assess the appropriateness of excluding data points from the burst pressure correlation on a case-by-case basis.

4.3.4 Combined Accident Loadings

Combined accident condition loadings such as loss of coolant accident (LOCA) plus safe shutdown earthquake (SSE) could result in yielding at a tube support plate (TSP) with subsequent deformation of the tubes. If significant tube deformation should occur, primary flow area could be reduced and postulated cracks in tubes could propagate through-wall resulting in the potential for in-leakage under LOCA conditions. In-leakage is a potential concern as leakage through several severed tubes may inhibit the core refill/reflood process and cause an unacceptable increase in the core peak clad temperature (PCT).

The most limiting accident conditions from tube deformation considerations are seismic (SSE) plus LOCA. The seismic excitation applied to steam generators is defined in the form of acceleration response spectra at the steam generator supports. In the seismic analysis, the licensee has used generic response spectra, which envelope the Kewaunee specific response spectra. A finite element model of the Series 51 steam generator was developed and the analysis was performed using the WECAN computer program. The mathematical model consisted of three dimensional lumped mass, beam, and pipe elements as well as general matrix input to represent the piping and support stiffnesses. Interactions at the TSP/shell and wrapper/shell connections were represented by concentric spring-gap dynamic elements. Impact damping was used to account for energy dissipation at these locations.

LOCA loads developed as a result of transient flow following a postulated primary coolant pipe break were calculated for five different pipe break locations. These included three large and two minor pipe breaks. The large pipe break locations evaluated were the steam generator inlet and outlet lines and the reactor coolant pump outlet line, while the minor pipe breaks analyzed were the pressurizer surge line and the accumulator line breaks. Prior qualification of the Kewaunee primary piping for leak before break requirements resulted in the limiting LOCA event being either the accumulator line break or the pressurizer surge line break. The licensee has however, used the loads for the primary piping break as a conservative approximation.

The principal tube loading from a LOCA is caused by the rarefaction wave in the primary fluid. This wave initiates at the postulated break location and travels around the tube U-bends. A differential pressure is created across the two legs of the tube, which causes an inplane horizontal motion of the U-bends and induces significant lateral loads on the tube. The pressure time histories needed for creating the differential pressure across the tube are obtained from transient thermal-hydraulic analyses using the MULTIFLEX computer code. For the rarefaction wave induced loadings, the predominant motion of the U-bends is along the plane of the U-bend. Thus, the individual tube motions are not coupled by the anti-vibration bars and the structural analysis is performed using single tube models limited to the U-bend and the straight leg region over the top two TSPs.

In addition to the rarefaction wave loading discussed above, the tube bundle is subjected to bending loads during a LOCA. These loads are due to the shaking of the steam generator caused by the break hydraulics and reactor coolant loop motion. However, the resulting TSP loads from this motion are small compared to those due to the rarefaction wave induced motion.

To obtain the LOCA induced hydraulic forcing functions, a dynamic blowdown analysis is performed to generate the system hydraulic forcing functions assuming an instantaneous double-ended guillotine break. The hydraulic forcing functions are then applied, along with the displacement time-history of the reactor pressure vessel (obtained from a separate reactor vessel blowdown analysis), to a system structural model, which includes the steam generator, the reactor coolant pump and the primary piping. This analysis yields the time history displacements of the steam generator at its upper lateral and lower support nodes. These time-history displacements formulate the forcing functions for obtaining the tube stresses due to LOCA shaking of the steam generator.

In calculating combined TSP loads, the LOCA rarefaction and LOCA shaking loads are combined directly, while the LOCA and SSE loads are combined using the square root of the sum of the squares. The overall TSP load is transferred to the steam generator shell through wedge groups located at discrete locations around the plate circumference.

The radial loads due to combined LOCA and SSE could potentially result in yielding in the TSP at the wedge support. Some tubes in the vicinity of the wedge supports could partially deform and subsequently collapse during a LOCA. The reduction in flow area increases the resistance to flow of steam from the core, which in turn may potentially increase PCT. In addition, there is a potential concern that partial through-wall cracks in a steam generator tube could progress to through-wall cracks during tube deformation. The resulting in-leakage is a potential concern since the cumulative leakage may cause an increase in the core PCT.

Utilizing results from recent tests and analysis programs, the licensee has shown that tubes will undergo permanent deformation if the change in diameter exceeds 0.025-inch. This threshold for tube deformation is related to the concern for tubes with preexisting through-wall cracks that could potentially open during a combined LOCA plus SSE event. For the Kewaunee plant, the LOCA plus SSE loads were determined to be of such magnitude that none of the tubes are predicted to exceed this deformation limit and therefore, will not be subjected to significant tube leakage.

The licensee has assessed the effect of SSE bending stresses on the burst strength of tubes with axial cracks. Tensile stress in the tube wall would tend to close the cracks while compressive stress would tend to open the cracks. On the basis of previously performed tests, the licensee has concluded that bending stresses on the order of yield stress of the tube material is necessary before the burst strength of the tube is affected to any significant degree. The maximum calculated bending stress in a tube wall during a seismic event is substantially less than the yield stress of the tube material. Thus, it is concluded that the burst strength of tubes with through-wall cracking is not affected by SSE event.

Based on a review of the information provided by the licensee for the Kewaunee plant, it is concluded that no significant tube leakage is likely to occur during and SSE plus LOCA event, which has been identified as the most limiting condition from tube deformation considerations.

4.4 Leakage Integrity

An important implication of voltage-based steam generator tube repair criteria is that the criteria may permit tubes to have, or to develop, through-wall or near through-wall cracks during the forthcoming operational cycle, thus creating the potential for primary-to-secondary leakage during normal operation, transients, or postulated accidents. Thus, the leakage integrity of these tubes, in addition to their structural integrity, must be assessed.

The staff finds that adequate leakage integrity during normal operating conditions is reasonably assured by the technical specification limits on allowable primary-to-secondary leakage. Adequate leakage integrity during transients and postulated accidents is demonstrated by showing that for the most limiting accident, assumed to occur at the end of the next operating cycle, the resulting leakage will not exceed a rate that will result in offsite dose limits being exceeded. The radiological consequences of this are discussed in Section 4.5.

4.4.1 Normal Operational Leakage

Implementation of the voltage-based tube repair criteria includes a reduction in the technical specification reactor coolant system leakage limits. Specifically, the present technical specification limit of 500 gallons per day (gpd) for primary-to-secondary leakage through any one steam generator is reduced to 150 gpd.

The present 500 gpd limit per steam generator is intended to ensure that through-wall cracks which leak at rates up to this limit during normal operation will not propagate and result in tube rupture under postulated accident conditions consistent with the criteria of Regulatory Guide 1.121. Development of the 150 gpd per steam generator leakage limit has utilized the extensive industry database regarding burst pressure as a function of crack length and leakage during normal operation. Based on leakage evaluated at the lower 95% confidence interval for a given crack size, the 150 gpd limit would be exceeded before the crack length reaches the critical crack length for MSLB pressures. Based on nominal, best estimate leakage rates, the 150 gpd limit would be exceeded before the crack length reaches the critical crack length corresponding to a burst pressure of three times normal operating pressure.

The reduced steam generator leakage limits to be adopted for implementation of the voltage-based tube repair criteria are more restrictive than the present operating leakage limits in the plant's technical specifications in order to provide a margin of safety against rupture. This reduction in the steam generator maximum allowable leakage limits is also intended to provide an additional margin in the event that a crack grows at a rate much greater than expected or which may unexpectedly extend outside the thickness of the tube support plate. The staff finds the proposed operating leakage limits in technical specification 3.1.d.2 to be acceptable for implementation of the voltage-based tube repair criteria.

4.4.2 Accident Leakage

The licensee has proposed a model for calculating the steam generator tube leakage from the faulted steam generator during a postulated MSLB which consists of two major components: (1) a model predicting the probability that a given indication will leak as a function of voltage (i.e., the probability of leakage (POL) model); and (2) a model predicting leak rate as a function of voltage, given that leakage occurs (i.e., the conditional leak rate model).

In the POL model, the probability that a given indication will leak is presented as a function of the bobbin coil voltage of that indication. The data is separated into two categories (i.e., indications which leak during a MSLB and those which do not). While various functional forms can be fitted to the data, the staff has concluded that a single functional form, the log-logistic, is acceptable for the purpose of assessing MSLB-induced steam generator tube leakage. The staff believes that any non-conservatism associated with the use of the log-logistic model, as compared to the other functional forms, is small compared to the conservatism inherent in the existing methodology for calculating the steam generator tube leakage and the radiological consequences of this leakage induced by a postulated MSLB. In addition, the differences in the POL functional forms are considered to be less significant when the leakage is calculated using a linear leak rate model, as discussed below, instead of a constant leak rate model which treats leakage as independent of voltage.

Regarding the conditional leak rate model, a correlation between the steam generator tube leak rate and bobbin voltage data based on a linear regression fit of the logarithms of the data has been developed. The staff provided statistical criteria in the draft generic letter on voltage-based repair criteria which permits licensees to use such a correlation if the correlation can be statistically justified at a 95% confidence level (i.e., a p-value of 5%). The staff concludes that using a linear relationship between the logarithms of the leak rate and bobbin voltage is appropriate in the determination of the primary-to-secondary steam generator tube leakage during a postulated MSLB provided the statistical criteria delineated in the draft generic letter on this subject are met. If the statistical criteria in the draft generic letter are not met, the linear regression should be assumed to have zero slope (i.e., the linear regression fit should be assumed to be constant with voltage). The staff further notes that the databases used in such evaluations should be consistent with the databases discussed in Section 4.4.3 of this evaluation.

The licensee has proposed a method for determining the primary-to-secondary steam generator tube leakage during a postulated MSLB which involves a Monte Carlo method which simulates the regression parameter uncertainties. The staff has analyzed this model for the case where the p-value test is valid at the 5% level, and has concluded that this model is appropriate and consistent with the draft generic letter on voltage-based repair criteria. This method involves:

1. Determining random versions of the POL and leak rate correlations to account for the uncertainty in the regression parameters (i.e., parameter uncertainty).

2. Using the regression parameters from Step 1 to determine the leak rate for each flaw indication in the estimated EOC voltage distribution. The EOC voltage distribution used in this calculation is the same as that discussed in Section 4.3.2.

3. Calculating the sum of the individual leak rates determined in Step 2 to obtain a value of the total steam generator leak rate.

4. Repeating Steps 1, 2, and 3 many times (e.g., 10,000) to obtain a distribution of the total steam generator leak rates.

5. Ordering the distribution of total leak rates in Step 4 in ascending order, and taking the 95th quantile at a 95% confidence level as the primary-to-secondary steam generator leakage during a postulated MSLB. This is the value used in assessing the leakage integrity of the steam generator tubing.

The staff notes that some minor variations in the details of the modeling may be necessary for the case where the p-value test is invalid at the 5% level.

The licensee has calculated the allowable steam generator leak rate to be 34.0 gallons per minute (gpm) in the faulted steam generator. This value is intended to be consistent with maintaining the radiological consequences of a release outside containment to within a small fraction of the guideline values in 10 CFR Part 100 as discussed in Section 4.5. As a result, if the primary-to-secondary leakage during a postulated MSLB is less than the 34.0 gpm limit, steam generator tubing affected by axially oriented ODSCC at the tube support plate elevations will maintain adequate leakage integrity under these conditions. The staff, therefore, finds this limit acceptable.

4.4.3 Data Exclusion from the Leakage Correlations

During the performance of the pulled tube examinations, malfunctions in the test equipment or improper specimen preparation can occasionally occur which could result in erroneous readings. Data such as this should not be included in the database since it could result in invalid results and/or conclusions. The staff, therefore, concluded in draft NUREG-1477 that eliminating data from the conditional leak rate and probability of leakage databases was appropriate provided that the data could be shown to be erroneous or the result of an invalid test. The staff provided additional guidance regarding the exclusion of data from the databases used in the steam generator tube leakage evaluation in a meeting with the industry on February 8, 1994. As a result of this guidance, the industry provided criteria for determining whether data may be removed from the probability of leakage and conditional leak rate databases. The specific criteria are presented in a letter referenced by the licensee which was submitted to the NRC by the Electric Power Research Institute (EPRI) on April 22, 1994.

The data points excluded from the conditional leak rate database and the probability of leakage database as a result of applying these criteria are listed in Tables E-2 and E-3 of the EPRI April 22, 1994, letter. The staff has concluded that excluding the data points listed in Table E-2, with the exception of model boiler specimen 542-4 and pulled tube specimen J1-R8C74, from the 7/8-inch conditional leak rate database; and excluding the data points listed in Table E-3 from the 7/8-inch diameter POL database is appropriate since it meets the exclusion criteria discussed by the staff at the February 8, 1994, industry meeting. Pending further evaluation of the generic criteria presented in Section E.2 of the April 22, 1994 letter, the staff is continuing to assess the appropriateness of excluding data points from the conditional leak rate and POL database on a case-by-case basis.

4.5 Assessment of Radiological Consequences

In support of the amendment request, the licensee presented its assessment of the radiological dose consequences of a 34 gpm primary to secondary leak initiated by a main steam line break accident. In the assessment, the licensee assumed that the allowable activity level of dose equivalent ^{131}I was 1.0 $\mu Ci/g$ for the primary coolant and 0.1 $\mu Ci/g$ for the secondary coolant. Two assessments were presented. One was based upon a preexisting iodine spike and the other was based upon an accident initiated iodine spike. The licensee presented doses for individuals located at the Exclusion Area Boundary (EAB) and at the Low-Population Zone (LPZ). The licensee concluded that, based upon a limit of 30 rem thyroid at the EAB, a leak rate of 34 gpm was determined to be the upper limit for allowable primary to secondary leakage in the steam generator in the faulted loop.

The staff independently calculated the doses resulting from a main steamline break accident using the methodology associated with Standard Review Plan (SRP) 15.1.5, Appendix A. The assumptions which were utilized by the staff in its calculations are presented in the Attachment. The results of the staff's calculations confirm the licensee's conclusions that the doses would be less than the limits established by SRP 15.1.5, Appendix A.

5.0 SUMMARY

Based on the above evaluation, the staff concludes that adequate structural and leakage integrity of the indications accepted for continued service under the voltage-based repair criteria can be ensured for Cycle 21 (1995 to 1996) at Kewaunee, consistent with applicable regulatory requirements. The staff's approval of the proposed voltage-based repair criteria is based, in part, on the licensee being able to demonstrate that the conditional probability of burst and the primary-to-secondary leakage during a postulated MSLB will be acceptable.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

This amendment changes 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 and changes surveillance requirements. 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 (59 FR 63127). 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.

8.0 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) 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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Date: April 17, 1995

Attachment: Input Parameters for Kewaunee Evaluation of Main SteamLine Break Accident

ATTACHMENT

INPUT PARAMETERS FOR KEWAUNEE EVALUATION OF MAIN STEAMLINE BREAK ACCIDENT

1. Primary coolant concentration of 60 μCi/g of dose equivalent ^{131}I.

   Preexisting Spike Value (μCi/g)

   ^{131}I = 32.7
   ^{132}I = 13.1
   ^{133}I = 53.8
   ^{134}I = 8.23
   ^{135}I = 29.5

2. Volume of primary coolant and secondary coolant.

Primary Coolant Volume (ft^3)	6236
Primary Coolant Temperature (°F)	578
Secondary Coolant Steam Volume (ft^3)	3838
Secondary Coolant Liquid Volume (ft^3)	1920
Secondary Coolant Steam Temperature (°F)	510.8
Secondary Coolant Feedwater Temperature (°F)	427.3

3. TS limits for DE ^{131}I in the primary and secondary coolant.

Primary Coolant DE ^{131}I concentration (μCi/g)	1.0
Secondary Coolant DE ^{131}I concentration (μCi/g)	0.1

4. TS value for the primary to secondary leak rate.

Primary to secondary leak rate, maximum any SG (gpd)	150
Primary to secondary leak rate, total all SGs (gpd)	150

5. Maximum primary to secondary leak rate to the faulted and intact SGs.

Faulted SG (gpm)	34
Intact SG (gpm)	0.1

6. Iodine Partition Factor

Faulted SG	1
Intact SG	0.1
Primary to Secondary Leakage	1.0

7. Steam Released to the environment

Faulted SG (lbs/2 hours)	99,300
Intact SG (lbs/2 hours)	209,000

8. Letdown Flow Rate (gpm) 40

9. Release Rate for 1 μCi/g of Dose Equivalent ^{131}I

		Ci/day
^{131}I	=	181
^{132}I	=	465
^{133}I	=	455
^{134}I	=	688
^{135}I	=	460

10. Atmospheric Dispersion Factors

EAB (0-2 hours)	2.9×10^{-4}
LPZ (0-8 hours)	5.2×10^{-5}