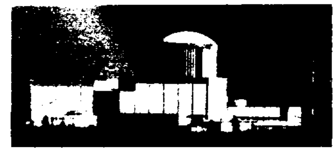
NRC-01-063

NMC
Committed to Nuclear Excellence

Kewaunee Nuclear Power Plant
N490, State Highway 42
Kewaunee, WI 54216-9511
920-388-2560

Operated by
Nuclear Management Company, LLC



May 25, 2001

10 CFR §50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

DOCKET 50-305
OPERATING LICENSE DPR-43
KEWAUNEE NUCLEAR POWER PLANT
PROPOSED AMENDMENT 177 TO KEWAUNEE NUCLEAR POWER PLANT
TECHNICAL SPECIFICATION 4.2

Pursuant to 10 CFR §50.90, Nuclear Management Company, LLC, (NMC) proposes to amend Kewaunee Nuclear Power Plant (KNPP) Facility Operating License DRP-43 by incorporating the attached changes into KNPP Technical Specifications (TS).

NMC intends to replace existing KNPP Westinghouse Model 51 original steam generators (OSG) with Westinghouse Model 54F replacement steam generators (RSG) in the fall of 2001, and is amending TS 4.2 as part of that replacement. The proposed change removes all NRC approved steam generator tube alternate repair criteria from TS 4.2 because these alternate repair criteria, as approved, are not compatible with RSG design.

Since removing alternate repair criteria requires a major revision to TS 4.2, NMC has taken this opportunity to make desired administrative changes. These changes revise the phrasing of text within the TS and are intended to simply enhance clarity of meaning and ease reader comprehension. They do not alter technical content or change the intended meaning of the affected text.

In accordance with 10 CFR §50.90, this letter seeks Nuclear Regulatory Commission (NRC) permission to make the foregoing changes to TS.

Nothing in this letter should be construed to constitute a commitment or redefine a margin of safety unless specifically so stated in separate correspondence or in a safety analysis of record.

In accordance with 10 CFR §50.30(b), this license amendment request is signed under oath or affirmation. Additionally, NMC has transmitted a copy of this license amendment request to the State of Wisconsin as required by 10 CFR §50.91(b)(1).

A001

If there are questions regarding this amendment, please contact either Mr. Thomas J. Webb at (920) 388-8537 or me at (920) 755-7627.

Sincerely,

Mark E. Reddemann
Site Vice President

Subscribed and Sworn to
Before Me This 25 Day
Of May 2001

Notary Public, State of Wisconsin

My Commission Expires
October 24, 2004

MTVN

Attachments:

1. Description of Change, Safety Evaluation, Significant Hazards Determination, and Statement of Environmental Considerations
2. Strike-Out Pages for Technical Specification 4.2 and Bases, Tables, List of Tables, Figure, List of Figures, Table of Contents
3. Revised Pages for Technical Specification 4.2 and Bases, Tables, List of Tables, Figure, List of Figures, Table of Contents

cc - US NRC Region III
US NRC Senior Resident Inspector
Electric Division, PSCW

ATTACHMENT 1

Letter from M. E. Reddemann (NMC)

To

Document Control Desk (NRC)

Dated

May 25, 2001

<u>Proposed Amendment 177</u>

Description of Proposed Changes

Safety Evaluation

Significant Hazards Determination

Environmental Consideration

Introduction

Nuclear Management Company, LLC (NMC) intends to replace Kewaunee Nuclear Power Plant's (KNPP) Westinghouse Model 51 original steam generators (OSG) with Westinghouse Model 54F replacement steam generators (RSG) commencing in the fall of 2001.

As a part of the initial Steam Generator In-Service-Inspection (ISI) Program, KNPP found many steam generator (SG) tubes to be unserviceable under the existing criteria. KNPP plugged these tubes. It soon became clear that the rate of plugging under existing criteria would require premature SG replacement. Since this was a generic problem, the industry developed methods for satisfactorily repairing SG tubes and the Nuclear Regulatory Commission (NRC) approved them. KNPP subsequently received NRC approval for use of these alternate repair methods, and associated criteria, to repair tubes and extend the unit's SG service life. However, these licensed criteria are not compatible with RSG design.

This request removes alternate repair criteria, along with their supporting figures, tables, and references from KNPP Technical Specification (TS) 4.2, "ASME Code Class In-Service Inspection and Testing." It also makes conforming changes to supporting tables, figures, the Table of Contents, the List of Tables, and the List of Figures.

Currently, the Basis for TS 4.2.b.4 references WCAP-7832, "Evaluation of Steam Generator Tube, Tube Sheet, and Divider Plate Under Combined LOCA Plus SSE Conditions." However, WCAP-7832 is applicable only to the original steam generators and must be replaced by Westinghouse WCAP-15325, "Regulatory guide 1.121 Analysis for the Kewaunee Replacement Steam Generators." WCAP-15325 describes the analysis to determine limits for RSG tubing, establishing minimum allowable tube wall thickness in accordance with guidelines of NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes."

WCAP-15325 determines that RSG tubes retain adequate strength with less remaining tube-wall thickness than was determined by WCAP-7832 to be required for OSG tubes. Hence, the Plugging Limit Criteria found in TS 4.2.b.4, which are based on WCAP-7832, are conservative and bound WCAP-15235 results. The existing Plugging Limit Criteria remain unchanged except to use WCAP-15325 as reference.

Since removal of alternate repair criteria requires extensive revision of TS 4.2 , NMC has taken this opportunity to make desired administrative changes to improve clarity of meaning, to aid reader comprehension, and to make formatting consistent with the remainder of KNPP TS. These changes do not alter the technical content or intent of TS 4.2.

Description of Change to TS 4.2, "ASME Code Class In-Service Inspection and Testing"

The proposed amendment:

- deletes Table 4.2-3, "Steam Generator Repaired Tube Inspection"
- deletes Figure TS 4.2-1, "Application of Plugging Limit for a Westinghouse Mechanical Sleeve"
- revises TS 4.2.b, "Steam Generator Tubes," to remove alternate repair criteria,
- revises the Bases for TS 4.2.b,
- revises Table 4.2-2, "Steam Generator Non-Repaired Tube Inspection,"
- revises the Table of Contents to reflect the changes to TS 4.2.b,
- revises the List of Tables to reflect the above changes,
- revises the List of Figures to reflect the above changes, and
- replaces reference to WCAP-7832 in the basis for TS 4.2.b.4 with WCAP-15325.

TS markup pages and amended pages contained in Attachments 2 and 3, respectively, show change detail.

Revisions described above are consistent with Regulatory Guide 1.83, "In-service Inspection of Pressurized Water Reactor Steam Generator Tubes."

Safety Evaluation for Proposed Change to TS 4.2

Technical Specification (TS) surveillance of equipment important to safety plays a significant role in assuring safe operation. TS 4.2, "ASME Code Class In-Service Inspection and Testing," contains a surveillance for assuring the ability of steam generator (SG) tubes to perform their design function.

Steam generator tubes serve multiple functions. Principal among these are:

(1) providing a path for transfer of heat from the reactor coolant to secondary system feedwater, and

(2) providing a major portion of the reactor coolant pressure boundary (RCPB).

System design relies on RCPB integrity to ensure retention of fission products within the primary system. SG tube design balances the mutually exclusive characteristics of thinner tube walls to improve heat transfer with thicker tube walls to improve strength.

SG tubes are subject to a variety of wear or damage mechanisms that degrade their ability to perform their design function. Some of these mechanisms create fissures that extend through the wall of a SG tube and allow primary to secondary leakage. This damage does not necessarily indicate that a tube is approaching its engineered strength limit nor does it necessarily allow release of significant amounts of fission products.

Due to the properties of metals used in manufacture of SG tubes, once a tube flaw exceeds a certain size, it could rapidly propagate to a size sufficient to pass the full flow capacity of the tube from the primary to the secondary system. This failure mechanism becomes more likely under stresses imposed by a steam generator tube rupture or main steam line break design basis accident. Once a measured tube flaw exceeds established limits, it must be repaired to restore its design strength or plugged to isolate it from the reactor coolant system.

The industry has developed methodologies to repair SG tubes and, thereby, extend SG service life. Several of these methodologies are currently licensed for use by KNPP and have been applied to allow tubes to remain in service that otherwise would have been required to be plugged. Plugging a tube is a more conservative measure that positively prevents reactor coolant flow through the tube to the secondary system during accident conditions.

Design of the Westinghouse Model 54F SG is not compatible with the currently licensed alternate repair methodologies. Deleting the less restrictive alternate repair options defaults to the more restrictive criteria and results in a lower threshold for plugging degraded tubes. Change from a less restrictive standard to a more restrictive standard is conservative and is bounded by the less restrictive standard. Deleting the currently licensed and less restrictive alternate repair criteria from TS is a change in the conservative direction and is bounded by existing design bases accident and transient analyses. Thus, deletion of alternate repair criteria from KNPP TS preserves design basis accident and transient assumptions, preserves effectiveness of accident mitigation systems, preserves currently licensed dose consequence and, hence, does not involve an unreviewed safety question.

Significant Hazards Determination for Proposed Change to TS 4.2

NMC reviewed the proposed change in accordance with provisions of 10 CFR §50.92 and determined that it creates no significant hazards. The proposed change does not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

   Changing the technical specification within limits of the bounding accident analyses cannot change the probability of an accident previously evaluated or the currently licensed radiological consequence predicted by the analyses of record. Removal of an allowance for alternate repair criteria defaults to the more conservative repair criteria of plugging degraded tubes. Thus, nothing in this proposal will cause an increase in the probability or consequence of an accident previously evaluated.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

   Removal of alternate repair criteria from TS leaves in its place the more conservative, more restrictive criteria for plugging degraded steam generator tubes. Plugging degraded steam generator tubes is a currently licensed repair methodology for KNPP, is consistent with current plant design bases, and does not adversely affect any fission product barrier, nor does it alter the safety function of safety significant systems, structures and components or their roles in accident prevention or mitigation. Currently licensed design basis accident and

transient analyses of record bound the effect of plugging tubes. Thus, this proposal does not create the possibility of a new or different kind of accident.

3) Involve a significant reduction in the margin of safety.

   The proposed change does not alter the manner in which Safety Limits, Limiting Safety System Setpoints, or Limiting Conditions for Operation are determined. It places TS 4.2 in a more conservative configuration than that previously approved for use by the NRC. It conforms to plant design bases, is consistent with current safety analyses, and limits actual plant operation within analyzed and licensed boundaries. Removal of reference to use of alternate repair criteria from TS 4.2 and its Bases leaves existing and more conservative criteria in place. Thus, changes proposed by this request do not involve a significant reduction in the margin of safety.

Environmental Considerations

This proposed amendment involves a change to the Technical Specifications. It does not modify any facility components located within the restricted area, as defined in 10 CFR §20. NMC has determined that the proposed amendment involves no significant hazards considerations and no significant change in the types of effluents that may be released offsite and that there is no significant increase in the individual or cumulative occupational radiation exposure. This proposed amendment accordingly 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 this proposed amendment.

ATTACHMENT 2

Letter from M. E. Reddemann (NMC)

To

Document Control Desk (NRC)

Dated

May 25, 2001

Proposed Amendment 177

Strike-Out Pages for Technical Specification

Table of Contents

TS ii

TS v

TS vi

TS 4.2-1 through TS 4.2-12

TS B4.2-1 through TS B4.2-8

Table TS 4.2-2

Table TS 4.2-3

Figure TS 4.2-1

Section		Title	Page

LIST OF TABLES

LIST OF FIGURES

Note:

[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

4.2 ASME CODE CLASS IN-SERVICE INSPECTION AND TESTING

APPLICABILITY

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

OBJECTIVE

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

SPECIFICATION

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

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

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

3. Surveillance testing of pressure isolation valves:

   a. Periodic leakage testing[1] on each valve listed in Table TS 3.1-2 shall be accomplished prior to entering the OPERATING mode after every time the plant is placed in the COLD SHUTDOWN condition for refueling, after each time the plant is placed in a COLD SHUTDOWN condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair, or replacement work is performed.

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

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

b. Steam Generator Tubes

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

Imperfection is ~~an exception to~~a deviation from the dimension, finish, or contour required by a design drawing or specification.

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

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

Degraded Tube means a tube ~~contains an imperfection~~containing degradation that is ≥ 20% of ~~the~~nominal wall thickness ~~caused by degradation~~.

Defect means an imperfection ~~of such severity that it exceeds the~~that violates criteria used to determine acceptability of a tube for continued use in operation ~~plugging limit. A tube containing a defect is defective~~.

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

Tube is ~~the~~a single hollow metal cylinder that is an element of an array of similar cylinders inside each steam generator, through which Reactor Coolant flows, and by which heat is transferred from the Reactor Coolant ~~System~~to the secondary system feedwater. Taken as a whole, steam generator tubes form a major portion of the reactor coolant pressure boundary ~~past the hot leg side of the tubesheet and before the cold leg side of the tubesheet~~.

Plugged Tube is a tube ~~intentionally~~that has been removed from service by installing a mechanical device in each end of the tube to seal the tube in a manner that isolates it from the reactor coolant system ~~plugging in the hot and cold legs because it is defective, or because its continued integrity could not be assured~~.

~~Repaired Tube is a tube that has been modified by tube repair methods described in TS 4.2.b.4.a to allow continued service consistent with plant Technical Specifications regarding allowable tube wall degradation, or to prevent further tube wall degradation. A tube without repairs is a nonrepaired tube. This definition does not apply to the portion of the tube below the F* or EF* distance provided the tube is not degraded (i.e., no detectable degradation permitted) within the F* distance for F* tubes and within the EF* distance for EF* tubes.~~

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

~~F* Distance is the distance of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tubesheet. The F* distance is equal to 1.11 inches (plus an allowance for NDE uncertainty) and is measured downward from the bottom of the uppermost roll transition. The F* distance applies to roll expanded regions below the midpoint of the tubesheet.~~

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

~~EF* Distance is the distance of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tubesheet. The EF* distance is equal to 1.51 inches (plus an allowance for NDE uncertainty) and is measured downward from the bottom of the uppermost roll transition. The EF* distance applies to roll expanded regions above the midpoint of the tubesheet.~~

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

1. Steam Generator Sample Selection and Inspection

   ~~The i~~In-service inspection of steam generators may be limited to one steam generator per inspection period on an ~~rotating~~alternating ~~schedule~~basis. The tubes shall be selected for inspection as ~~encompassing the number of tubes determined~~set forth in TS 4.2.b.2.a, provided ~~the~~that previous inspections indicated ~~that~~the two steam generators are performing in ~~a like~~an acceptably similar manner.

2. Steam Generator Tube Sample Selection and Inspection

~~The tubes selected for e~~Each in-service inspection~~ shall~~:

a. ~~Include~~Shall include a number of tubes that is at least equal to 3% of the total number of ~~nonrepaired~~non-plugged tubes~~;~~ contained in both steam generators~~; and 20% of the total number of repaired tubes in both steam generators. The tubes~~Tubes shall be selected for ~~these~~inspections ~~shall be selected~~on a random basis except as noted ~~below and~~in TS 4.2.b.2.b.

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

b. ~~Concentrate~~Shall concentrate the inspection by ~~selection of~~selecting 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~~Shall include all non-plugged tubes in which previous inspections revealed degradation ~~in excess of~~that exceeded 20% ~~degradation~~of nominal wall thickness. ~~The previously degraded tubes need only be inspected about~~For these tubes, only the area ~~of~~previously identified ~~degradation indication if~~as degraded must be inspected, unless their inspection is ~~not employed~~also performed to satisfy requirements of TS 4.2.b.2.a and TS 4.2.b.2.b above.

~~Implementation of the steam generator tube/tube support plate repair 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 the lowest cold-leg 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. In addition to the sample required in TS 4.2.b.2.a through TS 4.2.b.2.c, all tubes which have had the F*, or EF*, criteria applied will be inspected each in-service inspection in the uppermost tubesheet roll expanded region. These tubes may be excluded from TS 4.2.b.2.c provided the only previous wall penetration of >20% was located below the F* or EF* distance. F* and EF* tubes will be inspected for a minimum of 2 inches below the bottom of the uppermost roll transition. The results of F* or EF* tube inspections are not to be used as a basis for additional inspection per Table TS 4.2-2 or Table TS 4.2-3.~~

~~e. In addition to the sample required in TS 4.2.b.2.a through TS 4.2.b.2.c, all laser weld repaired sleeved tubes will be inspected at the first in-service inspection following the repair. Subsequent inspections will include a minimum sample size consistent with TS 4.2.b.2.a.~~

~~During the first in-service inspection and each subsequent in-service inspection, at least 20% of the laser weld repaired sleeved tubes will be inspected using an ultrasonic inspection technique. The laser weld repaired tubes inspected with the ultrasonic technique shall be selected on a random basis. Actions based on the results of the ultrasonic inspection shall be as described in Table TS 4.2-3.~~

~~f. In addition to the sample required in TS 4.2.b.2.a through TS 4.2.b.2.c, all Westinghouse mechanical hybrid expansion joint sleeves allowed to remain in service by the length criterion of TS 4.2.b.4.c shall be inspected during each in-service inspection in the upper joint region as depicted in Figure TS 4.2-1.~~

~~g~~d. ~~The~~May not require inspection of the full length of each tube during the second and third sample inspections ~~during each in-service inspection may be less than the full length of each tube by concentrating~~but may concentrate the inspection only on those ~~areas of the tubesheet array and on those~~portions of the tubes ~~where tubes with imperfections were~~previously found degraded.

~~h~~e. Shall perform a tube inspection on each selected tube. If the eddy current inspection probe ~~a tube does not permit the passage of the eddy current inspection probe~~will not pass through the entire length of a tube, ~~and through~~including the U-bend, it shall be so recorded and the tube shall be characterized as degraded. ~~this shall be recorded and a~~An adjacent tube shall also be inspected. ~~The tube which did not allow passage of the eddy current probe shall be considered degraded.~~

f. ~~The~~Shall classify sample inspection results ~~of each sample inspection shall be classified into~~as belonging to one of the following three categories~~. For non-repaired tubes~~, and actions shall accordingly be taken as described in Table TS 4.2-2. ~~For repaired tubes, actions shall be taken as described in Table TS 4.2-3.~~

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	Between 5% and 10% of the total tubes inspected are degraded tubes, or ~~O~~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~~For all inspections, previously degraded tubes must exhibit significant (>10%) ~~further~~added wall penetration~~s~~ to be included in the above percentage calculations.

3. Inspection ~~Frequencies~~Frequency

~~The above required i~~In-service inspection~~s~~ of steam generator tubes shall be performed at the following ~~frequencies~~intervals:

a. In-service inspections may be performed during refueling outages, but shall be performed at ~~refueling~~intervals not ~~more than~~to exceed 24 calendar months ~~after the previous inspection.~~, except that the inspection interval may be extended to a maximum of 40 months ~~If~~if:

1. two consecutive inspections following service under AVT conditions, not including the pre-service inspection, ~~result in all inspection results falling~~yield results that fall into the C-1 category~~;~~, or ~~if~~

2. two consecutive inspections demonstrate that previously ~~observed~~ documented degradation sites ~~has~~have not continued to deteriorate and no ~~additional~~new degradation ~~has occurred~~is found~~; the inspection interval may be extended to a maximum of once per 40 months~~.

b. If the result~~s~~ of ~~the~~a steam generator in-service inspection ~~of a steam generator~~ conducted in accordance with Table TS 4.2-2 falls into Category C-3, the inspection ~~frequency~~interval shall be ~~increased~~reduced to ~~at least once per~~20 months. ~~The increase in inspection frequency~~The 20 month interval shall apply until a subsequent inspection meets the conditions ~~specified in~~set forth in TS 4.2.b.3.a ~~and~~for extending the interval ~~can be extended~~to ~~a~~ 40~~-~~ months ~~period~~.

c. Additional, unscheduled in-service inspections of each steam generator shall be performed ~~on each steam generator~~using the criteria set forth in Table 4.2-2 for a "1st SAMPLE INSPECTION" ~~in accordance with the first sample inspection specified in Table TS 4.2-2~~during ~~the~~shutdowns ~~subsequent~~consequent 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.d, or

   2. A seismic ~~occurrence~~event having a magnitude greater than the Operating Basis Earthquake, or

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

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

d. If ~~the type of~~there is a significant change in steam generator chemistry ~~treatment~~control methodology ~~is changed significantly~~, the steam generators shall be operated at power for three months while using the new treatment and shall then be inspected ~~at~~during 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 or for degradation except as specified in TS 4.2.b.6 for tubesheet crevice region in which the F* and EF* criteria is applied.~~

~~a.~~ Any tube ~~which, upon inspection, exhibits~~with tube wall degradation of 50% or more shall be plugged ~~or repaired prior to~~before returning the steam generator to service. If significant general tube thinning occurs, this criterion ~~will be~~is reduced to 40% wall degradation. ~~Tube repair shall be in accordance with the methods described in the following:~~

~~WCAP-14685, Revision 4, "Laser Welded Repair of Hybrid Expansion Joint Sleeves for Kewaunee Nuclear Power Plant,"~~

~~WCAP-14685, Revision 2, Addendum 1, "Laser Welded Repair of Hybrid Expansion Joint Sleeves for Kewaunee Nuclear Power Plant Addendum 1: Evaluation of Weld Repaired HEJ Sleeved Tubes;"~~

~~WCAP-11643, "Kewaunee Steam Generator Sleeving Report (Mechanical Sleeves);"~~

~~CEN-629-P Revision 2, "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves;"~~

~~CEN-632-P Revision 0, "Repair of Kewaunee Steam Generator Tubes Using a Resleeving Technique;" or~~

~~WCAP-13088, Revision 3, "Westinghouse Series 44 and 51 Steam Generator Generic Sleeving Report" including Addendum 1 to Revision 4.~~

~~b. Any Westinghouse mechanical hybrid expansion joint (HEJ) sleeve which, upon inspection, exhibits wall degradation of 23% or more shall be plugged or repaired prior to returning the steam generator to service. Figure TS 4.2-1 depicts a Westinghouse HEJ sleeve.~~

~~c. For disposition of parent tube indications in the upper joint of Westinghouse HEJ sleeved tubes,* as depicted in Figure TS 4.2-1, the following requirements will apply:~~

~~1. HEJ sleeved tubes shall be inspected with a non-destructive examination technique capable of locating the bottom of the hardroll upper transition. HEJ sleeved tubes with circumferential parent tube indications located ≥ 0.92 inch (plus an allowance for NDE uncertainty) below the bottom of the hardroll upper transition, as measured on the inside of the sleeve, may remain in service.~~

~~2. HEJ sleeved tubes with circumferential parent tube indications located < 0.92 inch (plus an allowance for NDE uncertainty) from the bottom of the hardroll upper transition, as measured on the inside of the sleeve, shall be plugged or repaired prior to returning the steam generator to service.~~

~~3. HEJ sleeved tubes with axial parent tube indications located in the parent tube pressure boundary, as depicted in Figure TS 4.2-1, shall be plugged or repaired prior to returning the steam generator to service.~~

~~d. Any Combustion Engineering leak tight sleeve which, upon inspection, exhibits wall degradation 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.~~

~~e. Any Westinghouse laser welded sleeve which, upon inspection, exhibits wall degradation of 23% 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.~~

5. ~~Tube Support Plate Plugging Limit~~Deleted

~~The following criteria are used for the disposition of a steam generator tube for continued service that is experiencing predominantly axially oriented 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 ≤ the upper voltage repair limit, 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 > the upper voltage repair limit will be plugged or repaired.~~

~~*TS 4.2.b.4.c is applicable for operating cycles 23 and 24 only.~~

~~d. If an unscheduled mid-cycle inspection is performed, the following repair limits apply instead of TS 4.2.b.5.a, b and c. The mid-cycle repair limits are determined from the following equation:~~

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - 2.0)\left(\frac{CL - \Delta t}{CL}\right)$$

~~Where:~~

~~V_{MURL} =~~	~~mid-cycle upper voltage repair limit based on time into cycle~~
~~V_{SL} =~~	~~structural limit voltage~~
~~NDE =~~	~~95% cumulative probability allowance for NDE uncertainty~~
~~Gr =~~	~~average growth rate per cycle length~~
~~CL =~~	~~cycle length (time between scheduled inspections)~~
~~Δt =~~	~~length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented~~
~~V_{MLRL} =~~	~~mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle~~
~~V_{URL} =~~	~~upper voltage repair limit~~

~~Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.2.b.5.a, b and c.~~

~~NOTE: The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.~~

6. ~~F* and EF* Tubesheet Crevice Region Plugging Criteria~~Deleted

~~The following criteria are to be used for disposition or repair of steam generator tubes experiencing degradation in the tubesheet crevice region.~~

~~a. Tubes with indications of degradation within the roll expanded region below the midpoint of the tubesheet may remain in service provided the distance from the bottom of the uppermost roll transition to the tip of the crack is greater than 1.11 inches (plus an allowance for NDE uncertainty). This criteria is called the F* criteria and applies to the factory roll expansion, or to additional roll expansions formed as an extension of the original roll. Any degradation existing below the F* (plus an allowance for NDE uncertainty) is acceptable for continued service.~~

~~b. Indications of degradation not repairable by TS 4.2.b.6.a may be repaired using the EF* criteria. The EF* region is located a minimum of 4 inches below the top of the tubesheet, and is formed by an additional roll expansion of the tube in the originally unexpanded length. Tubes with indications of degradation within the EF* region may remain in service provided the distance from the bottom of the uppermost roll transition to the tip of the crack is greater than 1.51 inches (plus an allowance for NDE uncertainty). Any degradation existing below EF* (including uncertainty) is acceptable for continued service.~~

7. Reports

a. Following each in-service inspection of steam generator tubes~~, if there are any tubes requiring plugging~~during which tubes are plugged~~or repairing~~, the number of tubes plugged ~~or repaired~~shall be reported to the Commission within 30 days. ~~This report shall include the tubes for which the F* or EF* criteria were applied.~~

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

1. Number of tubes inspected and extent of inspection~~of tubes inspected~~.

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

3. Identification of tubes plugged.

~~4. Identification of tubes repaired.~~

c. ~~Results of~~If a steam generator tube inspection ~~which~~result falls into Category C-3, ~~require~~the Commission shall be promptly (within 4 hours) ~~notification~~notified ~~of the Commission consistent with~~according to requirements of 10 CFR 50.72(b)(2)(i). A Licensee Event Report shall then be filed with the Commission ~~written follow up report~~as described by Specification 4.2.b.7.a and~~shall be submitted to the Commission consistent with Specification 4.2.b.7.a, using the Licensee Event Report System to satisfy the intent of~~as set forth in 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 projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next 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 indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.~~

~~5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1 x 10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.~~

BASIS

~~The plant~~Kewaunee Nuclear Power Plant design was not ~~specifically~~designed to ~~meet the requirements of~~Section XI of the ASME Code; therefore, 100% compliance may not be practically ~~feasible~~achievable~~ or practical~~. However, the design process did consider access for in-service inspection ~~was considered during the design~~and made modifications ~~have been made~~within design limitations ~~where practical~~to ~~make provisions for~~provide maximum access~~ within the limits of the current plant design~~. ~~Where practical~~To the extent practical, ~~the~~NMC performs inspection of ASME Code Class 1, Class 2, Class 3, and Class MC components ~~is performed~~in accordance with Section XI of the ASME Code. If an inspection required by the ~~code~~Code ~~required inspection~~is impractical, ~~a~~ NMC requests Commission approval for ~~a~~deviation from the requirement~~ is submitted to the Commission for approval~~.

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

Technical Specification 4.2.b

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

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

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

Technical Specification 4.2.b.1

If the steam generators are ~~shown to be~~performing in ~~a like~~an adequately similar manner, it is appropriate to limit the inspection to one steam generator per inspection interval on ~~a rotating schedule~~an alternating basis. ~~Economic~~This offers economic savings as well as reduction~~s~~ ~~in~~of ~~personnel~~radiation exposure and outage duration~~can be realized~~.

Technical Specification 4.2.b.2

~~Periodic i~~Inspection of the steam generator tubes ~~allows~~provides evaluation of their service condition. ~~As o~~Operational experience has ~~become available it is evident~~shown that certain types of steam generators are susceptible to generic degradation mechanisms. ~~Site~~It has also revealed site-specific steam generator tube degradation mechanisms~~has also occurred throughout the industry~~. The Kewaunee inspection program ~~at Kewaunee is designed to identify~~assesses both generic and ~~site~~site-specific tube degradation~~s mechanisms~~.

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

Technical Specification 4.2.b.3

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

~~There are several operational occurrences or transients that~~Several operational events or transients ~~will~~require ~~subsequent~~consequent steam generator tube inspections. These inspections ~~are required as a result~~must be performed after occurrence of excessive primary-to-secondary leakage or after transients ~~imposing~~that impose large mechanical and thermal stresses on the tubes.

Technical Specification 4.2.b.4

~~Steam generator tubes found with less than the minimum wall thickness criteria determined by analysis, as described~~Procedures, calculations, and analyses found in WCAP-~~7832~~15325,[(1)~~(2)~~] combined with conservative allowances, such as general corrosion and measurement error, are the bases for the tube plugging criteria set forth in TS 4.2.b.4. Tubes that exceed the limits established by these criteria must ~~either~~be ~~repaired to be kept in service or~~removed from service by plugging.

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

~~To determine the basis for the sleeve plugging limit, the minimum sleeve wall thickness was calculated in accordance with the ASME Code and is consistent with Draft Regulatory Guide 1.121 (August 1976).~~

[(1)]WCAP ~~7832~~15325, "~~Evaluation of Steam Generator Tube, Tube Sheet, and Divider Plate Under Combined LOCA Plus SSE Conditions~~Regulatory Guide 1.121 Analysis for the Kewaunee Replacement Steam Generators."

~~[(2)]E. W. James, WPSC, to A. Schweneer, NRC, dated September 6, 1977.~~

~~For the Westinghouse mechanical hybrid expansion joint (HEJ) sleeves, the sleeve plugging limit of 23% is applied to the sleeve as shown on Figure TS 4.2-1. The sleeve plugging limits allow for eddy current testing inaccuracies and continued operational degradation per Draft Regulatory Guide 1.121 (August 1976).~~

~~Repair by sleeving, or other methods, has been recognized as a viable alternative for isolating unacceptable tube degradation and preventing tube leakage. Sleeving isolates unacceptable degradation and extends the service life of the tube, and the steam generator. Tube repair, by sleeving in accordance with WCAP-11643[3] and WCAP-13088,[4] has been evaluated and analyzed as acceptable. The Westinghouse mechanical HEJ sleeve spans the degraded area of the parent tube in the tubesheet region. The sleeves are either 36", 30" or 27" to allow access permitted by channel head bowl geometry. The sleeve is hydraulically expanded and hard rolled into the parent tubing.~~

~~The pressure boundary for HEJ sleeves is shown on Figure TS 4.2-1. The pressure boundary used to disposition parent tube indications (PTIs) detected in the upper joint of HEJ sleeved tubes is discussed in WCAP-15050.[5]* The pressure boundary described in the WCAP will allow PTIs located in the upper joint to remain in service if there is a minimum non-degraded (i.e., no detectable degradation in the parent tube) hardroll length of 0.92 inch (plus an allowance for NDE uncertainty) as measured from the bottom of the hardroll upper transition. The minimum hardroll engagement length is derived from structural and leakage testing. During field application, the PTI is located in reference to the bottom of the hardroll upper transition to ensure the minimum length of non-degraded hardroll exits. The inspection is performed using eddy current techniques capable of profiling and flaw detection as described in "NDE Technique to Determine Length Measurements in HEJ Sleeved Tubes with Parent Tube Indications."[6] The NDE uncertainty for this criterion is a function of the eddy current probe and technique used. The uncertainty has been calculated to be 0.023 inch. However, for field application, an eddy current uncertainty of 0.03 inch will be applied to the minimum hardroll engagement length of 0.92 inch.~~

~~[3]WCAP-11643, "Kewaunee Steam Generator Sleeving Report," Revision 1, November 1988 (Proprietary).~~

~~[4]WCAP-13088, Revision 3, "Westinghouse Series 44 and 51 Steam Generator Generic Sleeving Report," January 1994 (Proprietary) including Addendum 1 to Revision 4, June 1998 (Proprietary).~~

~~[5]WCAP-15050, "HEJ Sleeved Tube Length Based Degradation Acceptance Criterion," May 1998.~~

~~*The pressure boundary described by WCAP-15050 is applicable for operating cycles 23 and 24 only.~~

~~[6]"NDE Technique to Determine Length Measurements in HEJ Sleeved Tubes with Parent Tube Indications," Attachment 5 to Letter to Document Control Desk from C.R. Steinhardt dated May 14, 1998.~~

~~Leakage testing performed for the HEJ pressure boundary showed that leak rates for normal operating and steam line break (SLB) are comparable. However, statistical analysis shows that for a 99 percent confidence level the ratio of leak rate at SLB to normal operating is 9.3.[5] To bound SLB leak rate, the assumption is made that SLB leak rate is one order of magnitude greater than normal operating leak rate. The normal operating primary-to-secondary leakage limit is 0.104 gpm (150 gpd per TS 3.1.d.2). Therefore, the maximum primary-to-secondary leak rate during a SLB is assumed to be approximately 1 gpm (9.3 x 0.104 gpm). The 1 gpm will be the assigned leakage encompassing the HEJs left in service using the length criterion described in the paragraph above. Steam line break leakage in the faulted loop from all sources must be calculated to be less than or equal to the maximum allowable leakage described in the Basis for TS 3.4.d. Maintenance of the maximum allowable leak rate limit ensures off-site doses will remain within a small fraction of the 10 CFR Part 100 guidelines and ensures control room doses will not exceed GDC-19 during a SLB.~~

~~Recent inspection information has indicated a potential for the parent tube behind the upper HEJ region to develop service induced degradation. For parent tube degradation within or below the upper HEJ hardroll lower transition, tube operability can be restored by fusing the sleeve and tube using a laser welding process effectively isolating the degradation below the weld. The laser weld repair is performed similar to the initial installation of laser welded sleeves. The laser repair weld for degraded parent tubes with installed HEJ sleeves has been shown to meet the weld qualification, stress and fatigue requirements of the ASME code. All laser weld repaired HEJ sleeved tubes will receive a post weld stress relief at the weld location and ultrasonic inspection to verify weld quality, in accordance with the process described in WCAP-14685, Revision 4[7] and WCAP-14685, Revision 2, Addendum 1.[8]~~

~~[7]WCAP-14685, Revision 4, "Laser Welded Repair of Hybrid Expansion Joint Sleeves for Kewaunee Nuclear Power Plant," July 1998 (Proprietary).~~

~~[8]WCAP-14685, Revision 2, Addendum 1, "Laser Welded Repair of Hybrid Expansion Joint Sleeves for Kewaunee Nuclear Power Plant Addendum 1: Evaluation of Weld Repaired HEJ Sleeved Tubes," April 1997 (Proprietary).~~

~~Topical CEN-629-P[9] describes 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 a full depth tubesheet sleeve which is welded near the sleeve upper end and hard rolled into the tube and tubesheet at the sleeve lower end. A variation on the tubesheet sleeve design is the use of a pre-curved sleeve which allows access to the outer periphery of the tube bundle. 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. CEN-632-P[10] describes the steps required to re-sleeve tubes which have existing HEJ sleeves. This report describes the sleeved/tube preparation, re-sleeve installation and the design of a leak tight full depth tubesheet sleeve that is up to 39 inches in length.~~

~~Two types of Westinghouse laser welded sleeves can be installed, tube support plate sleeves and tubesheet sleeves.~~

~~The tube support plate sleeve is 12" long and spans the degraded area of the tube adjacent to the support plate intersection. The tube support plate sleeve is hydraulically expanded and laser welded at each end. The pressure boundary portion of the tube support plate sleeve is the weld and the sleeve section between the welds. Tubesheet sleeves extend from the tube end to above the top of the tubesheet. Standard and bowed or peripheral tubesheet sleeves can be installed. The upper or free span joint is hydraulically expanded and laser welded. The lower joint is hydraulically expanded and roll expanded. Standard tubesheet sleeves extend from 27" to 36" in length while bowed tubesheet sleeves extend from 30" to 36" in length. The pressure boundary portion of the tubesheet sleeve is the weld and below, down to the tubesheet primary face.~~

~~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.~~

~~[9] CEN-629-P Revision 2, "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves," January 1997.~~

~~[10] CEN-632-P Revision 0, "Repair of Kewaunee Steam Generator Tubes Using a Resleeving Technique," April 1997.~~

Technical Specification 4.2.b.5 (Deleted)

~~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 and GDC-19 limits are not exceeded.~~

~~The voltage-based repair limits of TS 4.2.b.5 implement the guidance in Generic Letter 95-05 and are applicable only to Westinghouse-designed steam generators with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The voltage-based repair limits are not applicable to other forms of tube degradation nor are they applicable to ODSCC that occurs at other locations within the steam generators. Additionally, the repair criteria apply only to indications where the degradation mechanism is predominantly axial ODSCC with no indications extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.~~

~~Implementation of TS 4.2.b.5 requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).~~

~~The voltage structural limit, V_{SL}, is the voltage from the burst pressure/bobbin voltage correlation, at the 95 percent prediction interval curve reduced to account for the lower 95/95 percent tolerance bound for tubing material properties at 650°F (i.e., the 95 percent LTL curve). The voltage structural limit must be adjusted downward to account for potential flaw growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit, V_{URL}, is determined from the structural voltage limit by applying the following equation:~~

$$\cancel{V_{URL} = V_{SL} - V_{GR} - V_{NDE}}$$

~~Where V_{GR} represents the allowance for flaw growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in GL 95-05.~~

~~The mid-cycle equation should only be used during unplanned inspection in which eddy current data is acquired for indications at the tube support plates.~~

Technical Specification 4.2.b.6 (Deleted)

~~Tubes with indications of degradation in either the original factory roll expansion in the tubesheet or the unexpanded portion of tube within the tubesheet may be dispositioned for continued service or repaired through application of the F* or EF* criteria. The F* and EF* criteria are described in WCAP-14677.[(11)] The F* and EF* criteria are established using guidance consistent with RG 1.121. Neither the F* or EF* criteria will significantly contribute to offsite dose following a postulated main steam line break such that contributions from these sources need to be included in offsite dose analyses. Inherent to these criteria is the ability to perform an additional roll expansion of the tube, either as an extension of the original factory roll expansion, in which case F* criteria applies, or in the area starting approximately 4 inches below the top of the tubesheet, in which case EF* criterion apply. The additional roll expansion procedure can be applied over existing degradation, provided the F* or EF* requirements for non-degraded roll expansion lengths of 1.11 inches (plus an allowance for NDE uncertainty) and 1.51 inches (plus an allowance for NDE uncertainty), respectively, are satisfied. The NDE uncertainty applied to the F* and EF* distance is a function of the eddy current probe and technique used. Current state-of-the art inspection technology will be used with implementation of the F* and EF* criteria. The uncertainty in such inspections has been shown to be as small as 0.06 inches, however, for field application, an eddy current uncertainty of 0.20 inches will be applied. Any and all indications of degradation existing below the F* or EF* distance is acceptable for continued service.~~

Technical Specification 4.2.b.7

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).

~~TS 4.2.b.7.d implements several reporting requirements recommended by GL 95-05 for situations which NRC wants to be notified prior to returning the steam generators to service. For TS 4.2.b.7.d.3 and 4, indications are applicable only where alternate plugging criteria is being applied. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle voltage distribution (refer to GL 95-05 for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the steam generators to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing GL Sections 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per GL Section 6.b(c) criteria.~~

~~[(11)]WCAP-14677, Revision 1, F* and Elevated F* Tube Alternate Repair Criteria for Tubes With Degradation Within the Tubesheet Region of the Kewaunee Steam Generators, May 1998 (Proprietary).~~

TABLE TS 4.2-2

STEAM GENERATOR ~~NON-REPAIRED~~ TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug ~~or repair~~ defective tubes and inspect additional 2S tubes in this S.G. (2)	C-1	None	N/A	N/A
			C-2	Plug ~~or repair~~ defective tubes and inspect additional 4S tubes in this S.G. (2)	C-1	None
					C-2	Plug ~~or repair~~ defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., (2) plug ~~or repair~~ defective tubes and inspect 2S tubes in the other S.G. (2) Prompt notification of the Commission. (1)	The other S.G.~~'s~~ ~~are~~is C-1	None	N/A	N/A
			~~Some~~Other S.G.~~'s~~ is C-2 ~~but no additional S.G. are C-3~~	Perform action for C-2 result of second sample	N/A	N/A
			~~Additional~~ Other S.G. is C-3	Inspect all tubes in ~~each~~other S.G. and plug ~~or repair~~ defective tubes. Prompt notification of the Commission. (1) (2)	N/A	N/A

S=6%/n Where n is the number of steam generators inspected during an inspection.

Notes:
1. Refer to Specification 4.2.b.7.c
2. As allowed by TS 4.2.b.2.~~g~~d, the second and third sample inspections during each inservice inspection may be less than the full length of each tube by concentrating the inspection on those ~~areas of the tube sheet array and on those~~ portions of the tubes where ~~tubes with~~ imperfections were previously found.

TABLE TS 4.2-3

Deleted

Deleted

ATTACHMENT 3

Letter from M. E. Reddemann (NMC)

To

Document Control Desk (NRC)

Dated

May 25, 2001

<u>Proposed Amendment 177</u>

Revised Pages for Technical Specification

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Note:

[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

4.2 ASME CODE CLASS IN-SERVICE INSPECTION AND TESTING

APPLICABILITY

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

OBJECTIVE

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

SPECIFICATION

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

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

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

3. Surveillance testing of pressure isolation valves:

   a. Periodic leakage testing[1] on each valve listed in Table TS 3.1-2 shall be accomplished prior to entering the OPERATING mode after every time the plant is placed in the COLD SHUTDOWN condition for refueling, after each time the plant is placed in a COLD SHUTDOWN condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair, or replacement work is performed.

---

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

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

b. Steam Generator Tubes

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

Imperfection is a deviation from the dimension, finish, or contour required by a design drawing or specification.

Degradation means service-induced cracking, wastage, wear or corrosion of a tube wall.

% Degradation is the amount in percent of tube wall thickness affected or removed by degradation.

Degraded Tube means a tube containing degradation that is $\geq$ 20% of nominal wall thickness.

Defect means an imperfection that violates criteria used to determine acceptability of a tube for continued use in operation.

Tube Inspection means the detailed examination of a steam generator tube from the point of entry (e.g., hot leg side) around the U-bend to the level of the top tube support plate of the opposite leg (cold leg).

Tube is a single hollow metal cylinder that is an element of an array of similar cylinders inside each steam generator, through which Reactor Coolant flows, and by which heat is transferred from the Reactor Coolant to the secondary system feedwater. Taken as a whole, steam generator tubes form a major portion of the reactor coolant pressure boundary.

Plugged Tube is a tube that has been removed from service by installing a mechanical device in each end of the tube to seal the tube in a manner that isolates it from the reactor coolant system.

1. Steam Generator Sample Selection and Inspection

In-service inspection of steam generators may be limited to one steam generator per inspection period on an alternating basis. The tubes shall be selected for inspection as set forth in TS 4.2.b.2.a, provided that previous inspections indicate the two steam generators are performing in an acceptably similar manner.

2. Steam Generator Tube Sample Selection and Inspection

Each in-service inspection:

a. Shall include a number of tubes that is at least equal to 3% of the total number of non-plugged tubes contained in both steam generators. Tubes shall be selected for inspection on a random basis except as noted in TS 4.2.b.2.b.

b. Shall concentrate the inspection by selecting 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. Shall include all non-plugged tubes in which previous inspections revealed degradation that exceeded 20% of nominal wall thickness. For these tubes, only the area previously identified as degraded must be inspected, unless their inspection is also performed to satisfy requirements of TS 4.2.b.2.a and TS 4.2.b.2.b above.

d. May not require inspection of the full length of each tube during the second and third sample inspections but may concentrate the inspection only on those portions of the tubes previously found degraded.

e. Shall perform a tube inspection on each selected tube. If the eddy current inspection probe will not pass through the entire length of a tube, including the U-bend, it shall be so recorded and the tube shall be characterized as degraded. An adjacent tube shall also be inspected.

f. Shall classify sample inspection results as belonging to one of the following three categories, and actions shall accordingly be taken as described in Table TS 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	Between 5% and 10% of the total tubes inspected are degraded tubes, or one or more tubes, but not more than 1% of the total tubes inspected, are defective.
C-3	More than 10% of the total tubes inspected are degraded tubes,or more than 1% of the inspected tubes are defective.
NOTE:	For all inspections, previously degraded tubes must exhibit significant (>10%) added wall penetration to be included in the above percentage calculations.

3. Inspection Frequency

In-service inspection of steam generator tubes shall be performed at the following intervals:

a. In-service inspections may be performed during refueling outages, but shall be performed at intervals not to exceed 24 calendar months, except that the inspection interval may be extended to a maximum of 40 months if:

   1. two consecutive inspections following service under AVT conditions, not including the pre-service inspection, yield results that fall into the C-1 category, or

   2. two consecutive inspections demonstrate that previously documented degradation sites have not continued to deteriorate and no new degradation is found.

b. If the result of a steam generator in-service inspection conducted in accordance with Table TS 4.2-2 falls into Category C-3, the inspection interval shall be reduced to 20 months. The 20 month interval shall apply until a subsequent inspection meets the conditions set forth in TS 4.2.b.3.a for extending the interval to 40 months.

c. Additional, unscheduled in-service inspections of each steam generator shall be performed using the criteria set forth in Table 4.2-2 for a "1st SAMPLE INSPECTION" during shutdowns consequent to:

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.d, or

2. A seismic event having a magnitude greater than the Operating Basis Earthquake, or

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

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

d. If there is a significant change in steam generator chemistry control methodology, the steam generators shall be operated at power for three months while using the new treatment and shall then be inspected during the next outage of sufficient duration.

4. Plugging Limit Criteria

Any tube with tube wall degradation of 50% or more shall be plugged before returning the steam generator to service. If significant general tube thinning occurs, this criterion is reduced to 40% wall degradation.

5. Deleted

6. Deleted

7. Reports

a. Following each in-service inspection of steam generator tubes during which tubes are plugged, the number of tubes plugged shall be reported to the Commission within 30 days.

b. The results of each steam generator tube in-service inspection shall be included in the Annual Operating Report for the reporting period that included completion of the inspection. The report shall include:

   1. Number of tubes inspected and extent of inspection.

   2. Location of each tube wall degradation and its percent of wall penetration.

   3. Identification of tubes plugged.

c. If a steam generator tube inspection result falls into Category C-3, the Commission shall be promptly (within 4 hours) notified according to requirements of 10 CFR 50.72(b)(2)(i). A Licensee Event Report shall then be filed with the Commission as described by Specification 4.2.b.7.a and as set forth in 10 CFR 50.73(a)(2)(ii).

BASIS

Kewaunee Nuclear Power Plant design was not designed to Section XI of the ASME Code; therefore, 100% compliance may not be practically achievable. However, the design process did consider access for in-service inspection, and made modifications within design limitations to provide maximum access. To the extent practical, NMC performs inspection of ASME Code Class 1, Class 2, Class 3, and Class MC components in accordance with Section XI of the ASME Code. If an inspection required by the Code is impractical, NMC requests Commission approval for deviation from the requirement.

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

Technical Specification 4.2.b

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

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

The NRC has developed guidance for steam generator tube inspection and maintenance including Regulatory Guides 1.83 and 1.121. Regulatory Guide 1.83, "In-service Inspection of Pressurized Water Reactor Steam Generator Tubes," forms the basis for many of the requirements in this section and should be consulted before revising them. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," defines steam generator tube minimum wall thickness.

Technical Specification 4.2.b.1

If the steam generators are performing in an adequately similar manner, it is appropriate to limit the inspection to one steam generator per inspection interval on an alternating basis. This offers economic savings as well as reduction of radiation exposure and outage duration.

Technical Specification 4.2.b.2

Inspection of the steam generator tubes provides evaluation of their service condition. Operational experience has shown that certain types of steam generators are susceptible to generic degradation mechanisms. It has also revealed site-specific steam generator tube degradation mechanisms. The Kewaunee inspection program assesses both generic and site-specific tube degradations.

Kewaunee uses various eddy current (EC) testing methodologies to inspect steam generator tubes. EC technology has improved considerably since Kewaunee began commercial operation in 1974, and NMC is committed to use advanced EC methods and technology, as appropriate, to assure accurate assessment of steam generator tube service condition.

Technical Specification 4.2.b.3

Kewaunee Nuclear Power Plant steam generator tube inspections are typically conducted during refueling outages. Criteria used to select tubes for inspection are based, in part, on tube service condition determined during previous inspections, and on operational experience from other plants with similar steam generators and water chemistry. Identification of degraded steam generator tubes results in expansion of the current inspection as well as increased frequency of subsequent inspections. In this manner, steam generator tube surveillance remains consistent with tube service condition.

Several operational events or transients require consequent steam generator tube inspections. These inspections must be performed after occurrence of excessive primary-to-secondary leakage or after transients that impose large mechanical and thermal stresses on the tubes.

Technical Specification 4.2.b.4

Procedures, calculations, and analyses found in WCAP-15325,[1] combined with conservative allowances, such as general corrosion and measurement error, are the bases for the tube plugging criteria set forth in TS 4.2.b.4. Tubes that exceed the limits established by these criteria must be removed from service by plugging.

Steam generator tube plugging is a common method of preventing excessive primary-to-secondary steam generator tube leakage. This method is relatively uncomplicated and isolates a defective tube from the reactor coolant system by installing mechanical devices to block its hot and cold leg tubesheet openings.

Technical Specification 4.2.b.5 (Deleted)

Technical Specification 4.2.b.6 (Deleted)

Technical Specification 4.2.b.7

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).

[1]WCAP 15325, "Regulatory Guide 1.121 Analysis for the Kewaunee Replacement Steam Generators."

TABLE TS 4.2-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G. (2)	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G. (2)	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., (2) plug defective tubes and inspect 2S tubes in the other S.G. (2) Prompt notification of the Commission. (1)	The other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in other S.G. and plug defective tubes. Prompt notification of the Commission. (1) (2)	N/A	N/A

S=6%/n Where n is the number of steam generators inspected during an inspection.

Notes:
1. Refer to Specification 4.2.b.7.c
2. As allowed by TS 4.2.b.2.d, the second and third sample inspections during each inservice inspection may be less than the full length of each tube by concentrating the inspection on those portions of the tubes where imperfections were previously found.

TABLE TS 4.2-3

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Figure TS 4.2-1