NMC
Committed to Nuclear Excellence
Kewaunee Nuclear Power Plant
Operated by Nuclear Management Company, LLC

NRC-03-102 10 CFR 50.90

October 8, 2003

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

KEWAUNEE NUCLEAR POWER PLANT
DOCKET 50-305
LICENSE No. DPR-43
LICENSE AMENDMENT REQUEST 199, "STEAM GENERATOR EDDY CURRENT INSPECTION FREQUENCY EXTENSION," TO THE KEWAUNEE NUCLEAR POWER PLANT TECHNICAL SPECIFICATIONS

The Nuclear Management Company, LLC, (NMC), in accordance with 10 CFR 50.90, is submitting this Licensing Amendment Request (LAR) to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS) to revise section 4.2.b.3.a "Inspection Frequency".

The proposed one-time change revises the Steam Generator (SG) inspection interval requirements in TS for KNPP to allow a 40-month inspection interval after one SG inspection. The reason for this one-time change is to eliminate unnecessary SG inspections, resulting in significant dose reduction and cost savings. We request approval of the proposed change prior to 6/30/2004. This would support postponing SG inspections during the fall 2004 refueling outage.

Attachment 1 to this letter contains a description, a safety evaluation, a significant hazards determination, and environmental considerations for the proposed changes. Attachment 2 contains the strike-out Technical Specification page TS 4.2-4. Attachment 3 contains the affected Technical Specification page as revised: TS 4.2-4.

Based on improved SG design, first in-service inspection results, and industry experience, NMC concludes that a second SG inspection is not required prior to proceeding with the 40-month inspection interval.

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The health and safety of the public will not be adversely affected by the proposed change. This submittal is not risk informed. This license amendment request is not needed for continued plant operation and does not contain proprietary information. Nothing in this LAR represents a commitment not previously made in a separate correspondence.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on October 8, 2003.

Thomas Coutu

Thomas Coutu
Site Vice President, Kewaunee Nuclear Power Plant
Nuclear Management Company, LLC

TLM

Attachments

cc: Administrator, Region III, USNRC
Senior Resident Inspector, Kewaunee, USNRC
Project Manager, Kewaunee, USNRC
Public Service Commission of Wisconsin

ATTACHMENT 1

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

October 8, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 199

Description of the Proposed Change

Safety Evaluation

Significant Hazards Determination

Environmental Considerations

Introduction

The proposed one-time change revises the Steam Generator (SG) inspection interval requirements in the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS) to allow a 40-month inspection interval after one SG inspection, rather than after two consecutive inspections resulting in C-1 category. KNPP TS define the C-1 category as < 5% of the total tubes inspected are degraded (i.e., contain defects greater than or equal to 20% throughwall) and none of the inspected tubes are defective (i.e., contain defects greater than or equal to the Plugging Limit Criteria of 50% or 40% if significant general tube thinning occurs.) The reason for this one-time change is to eliminate unnecessary SG inspections, resulting in significant dose reduction and cost savings.

We request approval of the proposed change prior to 6/30/2004. This would support eliminating SG inspections during the fall 2004 refueling outage. SG inspections will be performed during the spring 2006 refueling outage in accordance with the requirements in TS 4.2.b.2, "Steam Generator Tube Sample and Inspection", and Electric Power Research Institute (EPRI) "PWR Steam Generator Examination Guidelines," Volume 1, Revision 6, October 2002. The upcoming fall 2004 refueling outage is currently scheduled to begin October 9, 2004.

Description of Proposed Changes to Technical Specification TS 4.2.b.3.a, " Inspection Frequency "

A note will be added to TS 4.2.b.3.a, which modifies the exception to the Extension Criteria.

> NOTE: A one-time inspection interval extension of a maximum of once per 40 months is allowed following the inspection performed during the Spring 2003 Outage. This is an exception to the Extension Criteria in that the inspection interval extension is based on the result of only one inspection result falling into the C-1 category.

Safety Evaluation for Proposed Change to TS 4.2.b.3.a, " Inspection Frequency "

The inspection of the SG tubes ensures that the structural integrity of this portion of the Reactor Coolant System (RCS) will be maintained. As discussed below, the Spring 2003 refueling outage inspection results along with the improved Westinghouse replacement steam generator (RSG) design and industry operating experience regarding thermally treated Alloy 690 provides the basis for proposing an extension of the inspection interval while maintaining the integrity of the RCS. TS 4.2.b.3.a requires two consecutive inspections resulting in category C-1 classification before the inspection interval can be extended from a maximum of 24 calendar months to a maximum of 40 months. We are requesting that the inspection interval be extended to a maximum of 40 months on the basis of one SG inspection that resulted in all inspection results being classified in the C-1 category and improved SG design.

SG Design Improvements

Industry experience with recirculating SGs using mill annealed Inconel-600 tubing has led to significant design improvements in replacement SG design and fabrication. Problems associated with tube degradation (i.e., stress corrosion cracking (SCC), intergranular attack (IGA), pitting, and wastage) have been addressed through changes in tube materials and stress relief. Problems associated with secondary system fouling and flow-induced vibration and wear have been addressed with changes to the tube bundle support system. These design improvements, along with others, have been incorporated into the Westinghouse RSG design and are discussed below.

- Thermally Treated Inconel-690 Tubing

The KNPP RSGs are Westinghouse Model 54F SGs containing 3592 thermally treated (TT) Alloy 690 U-tubes with a nominal outer diameter of 0.875" and a thickness of 0.050". The development of thermally treated Inconel-690 tubing was prompted by the significant number of mill annealed Inconel-600 tubes being removed from service due to degradation.

Extensive testing has been performed which demonstrates thermally treated Inconel-690 tubing is superior to mill annealed Inconel-600 tubing in its resistance to both primary and secondary system SCC, pitting, and general corrosion. Examples of this data are given in EPRI Report 1003589, Pressurized Water Reactor General Tube Degradation Predictions (reference 1).

The tubing procurement specification used in construction of the Kewaunee Nuclear Plant, RSGs was designed to assure mill production of tubing that achieves the corrosion resistance properties as indicated by industry standards and research. The specification also outlines the physical, mechanical, and extensive inspection and qualification requirements necessary to limit fabrication defects.

In addition to the thermal treatment process that was performed on all tubing, additional stress relief was performed on all U-bends up to a 12" centerline radius (rows 1 through 8). The smallest centerline radius U-bend in the replacement SG design is 3.141". The additional stress relief on the low row U-bends was performed to relieve the residual stresses induced by the tube bending process. General experience with many tubing alloys indicates that high residual stresses associated with small radius bends increase the risk of stress corrosion cracking.

Industry inspection data supports the laboratory test results demonstrating the superior performance of thermally treated Inconel-690 as compared to mill annealed Inconel-600 tubing.

- Tube Bundle Support System

Experience with first generation mill annealed Inconel-600 tube recirculating SGs has identified the following issues relating to tube bundle support design.

- Dry-out and deposition in crevices of drilled hole type support plates leading to under deposit IGA.
- SCC resulting from denting of the tubes due to magnetite development on the carbon steel tube support plates or open crevices at the tubesheet joint.
- IGA and/or SCC associated with the high residual stress of rolled tube-to-tubesheet joint expansion particularly in combination with the unexpanded crevice design, which encouraged crevice corrosion as sludge concentrated in this critical area.
- Mechanical wear to the tube from fretting induced by flow-induced vibration, particularly in the U-bend area.

As described below, the Westinghouse model 54F replacement SG design incorporates features to greatly reduce or eliminate these potential damage mechanisms.

The RSG design uses seven, Type 405, stainless steel tube support plates with broached quatrefoil tube holes. The stainless steel construction of the tube support plates virtually eliminates the potential for denting. The quatrefoil tube holes also reduce the potential for dry-out and chemical concentration.

The Kewaunee SGs also contain a Type 405 stainless steel flow distribution baffle (FDB) located between the top of the tubesheet and the lowest tube support plate. The FDB is also largely open in the center. This increases the flow velocity across the tubesheet surface and locates the low flow velocity region (and sludge deposition zone) in the center of the tube bundle, near the blowdown intake. This reduces sludge accumulation and helps to mitigate corrosion.

Three sets of anti-vibration bar assemblies are installed in the U-bend region to stiffen the tube bundle, maintain proper tube spacing and alignment, and reduce tube vibration. The anti-vibration bars (AVBs) consist of V-shaped, rectangular cross section bars of Type 405 stainless steel material.

The tube-to-tubesheet joint accomplishes axial load resistance and the physical fastening of the tubing to the vessel. Original SG designs encountered severe corrosion problems with the SG tube-to-tubesheet joint region associated with open (i.e., unexpanded) crevices, and/or SCC at the high residual stress cold worked locations on the surface of the transition zone between the roll expanded and unexpanded tube. The RSG design incorporates the following features to address tube-to-tubesheet joint configuration concerns.

- Full depth expansion to eliminate a tubesheet crevice that could encourage denting or accumulation of contaminants against the transition zone.
- Hydraulic expansion that leaves minimal residual stresses and cold work as compared to mechanical roll expansion techniques. Hydraulic expansions typically produce 20 - 40% less stress than hard-rolled expansions.

First Outage Inspection Sampling

Kewaunee conducted the first inservice inspection on the replacement steam generators during the Spring 2003 refueling outage. Kewaunee performed significantly more than the minimum TS requirement during the spring 2003 refueling outage by inspecting:

- 100% full-length in both SGs with bobbin probes.
- 20% sample of Row 1 U-bends in both SGs with Plus Point probes adapted for U-bend use.
- 20% Hot Leg Top of tubesheet (H/L TTS) ±2 inches in both SGs with Plus Point probes biased towards peripheral tube locations.
- 100% of bobbin "I codes," dings/dents 5 volts or greater (special interest) and Possible Loose Parts (PLPs) with Plus Point Rotating Pancake Coil (RPC) probe.

The inspections were performed in accordance with the sampling requirements contained in Section 3.3.1, "Examination of Tubes," of the EPRI Pressurized Water Reactor (PWR) SG Examination Guidelines, revision 5. The inspection was performed using techniques qualified in accordance with the EPRI PWR SG Examination Guidelines Appendix H, "Performance Demonstration for Eddy Current Examination," and data analysis personnel qualified in accordance with the EPRI PWR SG examination guidelines, Appendix G, "Qualification of Nondestructive Examination Personnel for Analysis of NDE Data. The inspection results showed no degraded or defective tubes, and both SG were classified as category C-1. A summary of the eddy current examinations performed during the spring 2003 refueling outage was previously provided to the staff (reference letter NRC-03-082) as part of the 2003 ISI summary report.

The first outage inspection sampling results, along with industry experience and the operational assessment discussed below indicate that tube integrity will be maintained over the proposed operating period.

EPRI PWR SG Examination Guidelines

The EPRI PWR SG Examination Guidelines base SG inspection frequency on inspection results and performance criteria. KNPP has followed the recommendations of Section 3.3.1 of the ERPI PWR SG Examination Guidelines, revision 5, which contain the provisions regarding SG inspection frequency based on inspection results.

1. After the first cycle of operation (i.e., a duration not less than six effective full power months (EFPM) and not more than 24 EFPM for either new or replacement SGs, a 100% full-length (i.e., tube end to tube end) examination using general purpose eddy current probes shall be performed on all SGs.

2. During subsequent in-service inspections, if tube degradation (i.e., active damage mechanisms as defined in Appendix F. "Terminology," of the ERPI PWR SG Examination Guidelines) are identified, all SGs shall be examined at the end of each fuel cycle or 24 EFPM, whichever is less, or as necessary to satisfy published regulatory requirements.

3. During subsequent in-service inspections, if active damage mechanisms are not identified, the number of SGs to be examined and/or the frequency of examination shall be performed as required by Section 3.3.2, "Steam Generators Free from Active Damage Mechanisms," of the EPRI PWR SG Examination Guidelines.

4. 100% of the tubing and 100% of each type of repair shall be inspected within a rolling 60 EFPM time frame. If 60 EFPM occur during an operating cycle, completion of that cycle is acceptable and is within stated requirement.

5. No SG shall operate more than two fuel cycles between inspections.

KNPP is in the process of implementing revision 6 of the EPRI PWR SG Examination Guidelines, which contain specific inspection requirements for SGs tubed with thermally treated alloy 690 material. Revision 6 allows extension of the inspection frequency up to 144 months to complete a 100% inspection following the first in-service inspection when no active degradation mechanisms are present.

As stated in Section 3.3.2 of the ERPI PWR SG Examination Guidelines, revision 5, if the SGs are free from active damage mechanisms, some latitude is provided in terms of the number of SGs to be inspected and/or frequency of inspection. For these SGs, any of the following options may be performed.

1. Inspect ≥ 20% of the tubes and ≥ 20% of each type of repair in each SG at each refueling outage, or

2. Inspect ≥ 40% of the tubes and ≥ 40% of each type of repair in half the number of SGs at each refueling outage, or

3. Inspect ≥ 40% of the tubes and ≥ 40% of each type of repair in each SG at every other refueling outage.

Kewaunee Nuclear Plant does not have an active SG damage mechanism as defined in the ERPI PWR SG Examination Guidelines. Therefore, Option 3 as indicated above will be met without performing SG inspections during the upcoming Fall 2004 refueling outage.

Inspection frequency in accordance with Section 5.0 "Steam Generator Assessments," of the EPRI PWR SG Examination Guidelines is also determined by measuring results against performance criteria. As part of the KNPP SG Tube Surveillance Program, both a Condition Monitoring Assessment and Operational Assessment are performed after each inspection and the results are compared to performance criteria. The performance criteria against which the results are compared include SG tube structural integrity, accident-induced leakage, and operational leakage. The performance criteria are contained in Section 2.0 "Performance Criteria," of Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines" (Reference 4). The Condition Monitoring Assessment is retrospective and is intended to confirm that adequate SG integrity has been maintained since the previous inspection. The Operational Assessment is predictive and is intended to provide reasonable assurance that the performance criteria will be met throughout the next operating period. If it is determined that the performance criteria will not be met at the end of the operating period, then the operational cycle length must be adjusted accordingly.

Condition Monitoring Assessment

After completion of the Kewaunee Nuclear Plant, Spring 2003 refueling outage SG inspections, a Condition Monitoring Assessment was performed in accordance with EPRI "Steam Generator Integrity Assessment Guidelines," Revision 1, March 2000. This document provides guidelines for evaluating the condition of SG tubes based on inspection results. No tube degradation was detected in either SG and the condition monitoring assessment concluded that all performance criteria had been met.

Operational Assessment

An Operational Assessment was performed in accordance with EPRI SG Integrity Assessment Guidelines to evaluate the predicted condition of the SGs after two cycles of operation. The Operational Assessment is summarized below.

The operational assessment evaluated Kewaunee Nuclear Plant for an operating period lasting until the fall 2006 refueling outage. Both cycles were estimated to be approximately 1.5 effective full power years (EFPY), therefore the operational assessment assumed 3.0 EFPY of operation.

No tube degradation due to wear or corrosion was detected in either of the Kewaunee SGs during the 2003 refueling outage. Possible Loose Parts (PLPs) were detected by eddy current in both SGs. All PLPs were re-tested using RPC technology and there was no wear associated with any of the PLP indications. None of the PLPs in SG 1A were confirmed by visual exam indicating that the signals could be indications of local sludge deposits. Three objects were confirmed by visual exam in SG 1B.

Of the three objects confirmed by visual examination, one small machine chip was removed. Westinghouse performed a bounding analysis of the remaining two foreign objects (Reference 3). The largest of the foreign objects measured 1/8" to 1/4" in width, approximately 1/64" thick and had sufficient curl to permit lodging between two adjacent tubes. Hence a length of approximately 1" was used in the analysis. The analysis assumed an operating cycle of 3 years and determined that the energy which the loose object would impart to a tube during repeated collisions is very small and would result in a dent size of about 0.2 mils. Thus, based on experience with similar foreign objects and specific test data contained in reference 2, and the Westinghouse analysis described above, it is concluded that the identified foreign objects are not capable of causing significant damage to the Kewaunee steam generator tubes.

The operational assessment demonstrated that there is reasonable assurance that the structural and leakage limits will continue to be satisfied at the end of the next two operating cycles. The operational assessment of both SGs complies with the guidance of NEI 97-06 Revision 1, and demonstrates that the Kewaunee SGs are expected to continue to meet the leakage and structural integrity performance criteria for the duration of Cycles 26 and 27.

Industry Experience

The degradation assessment performed for the Spring 2003 outage addresses industry operating experience with Alloy 690 TT tubing. The most relevant operating experience with which to compare the Kewaunee SGs includes those generators with the same tubing material and generally the same design features. Comparisons are made with SGs that include similar design features and utilize Alloy 690 TT tubing to predict the anticipated performance of the Alloy 690 TT tubing. There are currently at least 70 operational plants world-wide employing SGs tubed with Alloy 690TT that have accrued approximately 400 reactor years service. In addition to Kewaunee, 7 other plants with comparable design RSGs have operated with Alloy 690 TT tubing since 1988. The service time on these SGs is listed in Table 1. Other RSGs manufactured by Westinghouse using Alloy 690TT tubes include the Delta series SGs which are also listed in Table 1. Only Sizewell B, employing Alloy 690TT tubing in Model F SGs, was so equipped as original equipment. The total operating experience for Westinghouse SGs with Alloy 690TT tubing is approximately 258 SG-years as of the middle of 2002.

Non-Westinghouse Alloy 690TT service - approximately 850 SG-years - has also accrued in plants with SGs supplied by Framatome (22 plants - 16 with RSGs), by Mitsubishi Heavy Industries (MHI) (16 plants - 11 with RSGs), by Siemens/Kraftwerk Union (KWU) (3 plants, all with RSGs) and by Babcock & Wilcox International (BWI) (9 plants, all with RSGs). Operations at these stations have accrued since as early as 1989. To date only wear related degradation mechanisms have been reported for these plants. Since about 1983, Alloy 690 sleeves have been used to repair plants affected by pitting and/or outside diameter stress corrosion cracking (ODSCC). While the total operating experience with Alloy 690 is limited compared to that of Alloy 600, it has been essentially flawless to date.

Table 1: Alloy 690TT Experience in Westinghouse SGs (6-1-02)

Plant	Commercial Operation	Replacement Date	Model	Tubes	Thot	SG-years*
COOK 2	7/1/78	3/1/89	54F	3592	606	53.00
FARLEY 1	12/1/77	5/24/00	54F	3592	607	6.06
FARLEY 2	7/1/81	5/8/01	54F	3592	607	3.20
INDIAN POINT 3	8/1/76	6/1/89	44F	3214	593	52.00
KEWAUNEE	6/1/74	12/5/01	54F	3592	592	0.97
MIHAMA 1	11/28/70	4/3/96	35F	2918	603	12.32
NORTH ANNA 1	6/1/78	4/1/93	54F	3592	613	27.50
NORTH ANNA 2	12/1/80	6/1/95	54F	3592	613	21.00
ANO 2	11/1/80	9/15/01	Delta 109	10637	603	1.42
KORI 1	4/29/78	7/1/98	Delta 60	4934	607	7.84
POINT BEACH 2	10/1/72	3/1/97	Delta 47	3499	597	10.50
SHEARON HARRIS	5/1/87	10/1/01	Delta75	6307	619	2.00
SOUTH TEXAS 1	8/25/88	5/1/00	Delta 94	7585	620	8.33
SOUTH TEXAS 2	4/11/89	10/02	Delta 94	7585	620	0.3
SUMMER	1/1/84	12/1/94	Delta75	6307	619	22.50
SIZEWELL B	2/1/95	Original in Service	F	5626	613	29.32

*: Elapsed time since SG placed in service x number of SGs per plant.

Dose and Cost Impact

If this proposed change is not approved, our current plan would be to perform 100% full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) bobbin inspection of both SGs during the Fall 2004 refueling outage. Assuming this scope, the following dose and cost impacts are predicted based on the Kewaunee Spring 2003 refueling outage.

- Accumulated personnel dose including SG platform setup, manway removal and eddy current inspection is estimated to be approximately 7 person-REM.

- The approximate cost associated with inspecting both SGs including contractor craft support is $900,000.

Based on improved SG design, first in-service inspection results, and industry experience, we have concluded that SG inspections in the fall 2004 outage are not necessary to meet the SG inspection objectives.

Regulatory Basis

Precedence has been set for this TS change when the NRC approved TS changes for Braidwood Station (TAC No. MB 1226 and 1227), South Texas Project (TAC No. MB 3963), and Farley Nuclear Plant, Unit 1 (TAC No. MB 4310). Each of these plants has utilized SGs manufactured with thermally treated Alloy 690. The NRC found that the safety performance had been significantly improved in the RSGs after the incorporation of the material changes and design changes. Each of these plants was allowed to go to a 40-month inspection cycle based on the first in-service inspection after SGR without another conservative inspection.

Significant Hazards Determination for Proposed Change to TS 4.2.b.3.a, " Inspection Frequency "

The proposed one-time change revises the Steam Generator (SG) inspection interval requirements in TS for the Kewaunee Nuclear Power Plant (KNPP) to allow a 40-month inspection interval after one SG inspection, rather than after two consecutive inspections resulting in C-1 category. C-1 category is defined as "< 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective". The reason for this one-time change is to eliminate unnecessary SG inspections, resulting in significant dose, schedule and cost savings.

The proposed changes were reviewed in accordance with the provisions of 10 CFR 50.92 to determine that no significant hazards exist. The proposed changes will not:

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

The proposed one-time change revises the Steam Generator (SG) inspection interval requirements in Technical Specifications (TS) 4.2.b.3.a, following the Kewaunee Nuclear Plant, spring 2003 refueling outage, to allow a 40-month inspection frequency after one inspection, rather than after two consecutive inspections results that are within the C-1 category.

The proposed one-time extension of the SG tube in-service inspection interval does not involve changing any structure, system, or component, or affect reactor operations. It is not an initiator of an accident and does not change any existing safety analysis previously analyzed in the Kewaunee Updated Safety Analysis Report (USAR). As such, the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

Since the proposed change does not alter the plant design, there is no direct increase in SG leakage. Industry experience indicates that the probability of increased SG tube degradation would be very low. Additionally, steps described below will further minimize the risk associated with this extension. For example, the scope of inspections performed during the last KNPP refueling outage (i.e., the first refueling outage following Steam generator replacement (SGR) exceeded the TS requirements for the first two refueling outages after SGR. That is, more tubes were inspected than were required by TS(i.e 100% inspection was performed). Currently, KNPP does not have an active SG damage mechanism, and will meet the current industry examination guidelines without performing additional SG inspections until the spring 2006 refueling outage. Additionally, as part of our SG Tube Surveillance Program, both a Condition Monitoring Assessment and an Operational Assessment are performed after each inspection and compared to the Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," performance criteria. The results of the Condition Monitoring Assessment demonstrated that all performance criteria were met during the KNPP spring 2003 refueling outage, and the results of the Operational Assessment show that all performance criteria will be met over the proposed operating period. Considering these actions, along with the improved SG design and reliability of Westinghouse replacement SGs, extending the SG tube inspection frequency does not involve a significant increase in the consequences of an accident previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change revises the SG inspection frequency requirements in TS 4.2.b.3.a, to allow a 40-month inspection interval after one inspection, rather than after two consecutive inspections with inspection results within the C-1 category.

The proposed change will not alter any plant design basis or postulated accident resulting from potential SG tube degradation. The scope of inspections (i.e. 100%) performed during the last KNPP refueling outage (i.e., the first refueling outage following SG replacement) significantly exceeded the TS requirements for the scope of the first two refueling outages after SG replacement.

Primary to secondary leakage that may be experienced during all plant conditions is expected to remain within current accident analysis assumptions. The proposed change does not affect the design of the SGs, the method of SG operation, or reactor coolant chemistry controls. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. The proposed change involves a one- time extension to the SG tube in-service inspection frequency, and therefore will not give rise to new failure modes. In addition, the proposed change does not impact any other plant system or components.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety?

The SG tubes are an integral part of the Reactor Coolant System (RCS) pressure boundary that are relied upon to maintain the RCS pressure and inventory. The SG tubes isolate the radioactive fission products in the reactor coolant from the secondary system. The safety function of the SG is maintained by ensuring the integrity of the SG tubes. In addition, the SG tubes comprise the heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system.

SG tube integrity is a function of the design, environment, and current physical condition. Extending the SG tube inservice inspection frequency by one operating cycle will not alter the function or design of the SG. SG inspections conducted during the first refueling outage following SG replacement demonstrated that the SGs do not have an active damage mechanism, and the scope of those inspections significantly exceeded those required by the TS. These inspection results were comparable to similar inspection results for similar replacement SGs installed at other plants, and subsequent inspections at those plants yielded results that support this extension request. The improved design of the replacement SGs also provides reasonable assurance that significant tube degradation is not likely to occur over the proposed operating period.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Environmental Considerations

The NMC has determined that the proposed amendment involves no significant hazard considerations. There are no changes in the types of any effluents that may be released off-site and that there are no increases in the individual or cumulative occupational radiation exposure. Accordingly, this proposed 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 this proposed amendment.

REFERENCES

1. EPRI Report 1003589, "Pressurized Water Reactor Generic Tube Degradation Predictions."

2. Combustion Engineering Report No. CENC 1278, "Investigation of the Effects of a Tube Guide in a San Onofre Steam Generator," December 10, 1976.

3. Westinghouse Letter LTR-SGDA-03-94, "Loose Parts Evaluation for Kewaunee (54F) for 1R26 2003 Spring Outage," April 28, 2003.

4. NEI 97-06, "Steam Generator Program Guidelines, draft Revision 1, dated December 11, 2000.

ATTACHMENT 2

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

October 8, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 199

Strike Out TS Page:

TS 4.2-4

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.

NOTE: A one-time inspection interval extension of a maximum of once per 40 months is allowed following the inspection performed during the spring 2003 inspection. This is an exception to the Extension Criteria in that the inspection interval extension is based on the result of only one inspection result falling into the C-1 category.

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.

ATTACHMENT 3

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

October 8, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

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Affected TS Page:

TS 4.2-4

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   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.

   NOTE: A one-time inspection interval extension of a maximum of once per 40 months is allowed following the inspection performed during the spring 2003 inspection. This is an exception to the Extension Criteria in that the inspection interval extension is based on the result of only one inspection result falling into the C-1 category

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.