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FITZPATRICK,E. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
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SUBJECT: Application for amend to license DPR-58,incorporating 2.0 volt interim SG tube support plate plugging criterion for fuel cycle 15.

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April 25, 1995 AEP:NRC:1166R

Docket Nos.: 50-315

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

Gentlemen:

Donald C. Cook Nuclear Plant Unit 1
TECHNICAL SPECIFICATION CHANGES TO INCORPORATE
2.0 VOLT INTERIM STEAM GENERATOR TUBE SUPPORT PLATE
PLUGGING CRITERION FOR FUEL CYCLE 15

This letter and its attachments provide supplemental information in response to the March 15, 1995, telephone discussions with your staff regarding our February 3, 1995, letter, AEP:NRC:1166Q, concerning application for amendment to the technical specifications (T/Ss) of Donald C. Cook Nuclear Plant Unit 1. Specifically, this supplemental information addresses the applicable requirements of draft NRC Generic Letter (GL) 94-XX, "Voltage - Based Repair Criteria for the Repair of Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

For your convenience, all attachments previously submitted with AEP:NRC:1166Q are being re-submitted and supersede those contained in that submittal. Attachment 1 provides a technical summary of the specific inspection practices and calculational methodologies outlined in GL 94-XX that will be applied to the Unit 1 interim plugging criteria program and the 10 CFR 50.92 no significant hazards evaluation. The evaluation and results support continued use of 2 volt interim plugging criteria for fuel cycle 15. Attachment 2 contains existing T/S pages marked to reflect the requested changes. Attachment 3 provides the proposed revised T/S pages.

A001

We believe the proposed changes will not result in (1) a significant change in the types of any effluent that may be released offsite, or (2) a significant increase in individual or cumulative occupational radiation exposure.

These proposed changes have been reviewed by the Plant Nuclear Safety Review Committee and will be reviewed by the Nuclear Safety and Design Review Committee at the next regularly scheduled meeting.

In compliance with the requirements of 10 CFR 50.91(b)(1), copies of this letter and its attachments have been transmitted to the Michigan Public Service Commission and to the Michigan Department of Public Health.

The T/S pages affected by those proposed changes are also impacted by the T/S pages submitted with our April 13, 1995, letter AEP:NRC:1129D, Use of Laser Welded Sleeves for Steam Generator Tubes.

Sincerely,

for E. E. Fitzpatrick
Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 25th DAY OF April 1995

Notary Public

My Commission Expires: 6-28-99

eh

Attachments

cc: A. A. Blind
G. Charnoff
J. B. Martin
NFEM Section Chief
NRC Resident Inspector - Bridgman
J. R. Padgett

ATTACHMENT 1 TO AEP:NRC:1166R

DESCRIPTION OF CHANGES
TO THE DONALD C. COOK NUCLEAR PLANT
UNIT 1 TECHNICAL SPECIFICATIONS

10 CFR 50.92 EVALUATION

I. INTRODUCTION

This amendment request proposes a change to Cook Nuclear Plant Unit 1 steam generators (SG) T/Ss 4.4.5.2, 4.4.5.3, 4.4.5.4, 4.4.5.5, 3.4.6.2 and Bases 3/4.4.5 and 3/4.6.2 to allow continued use of SG tube support plate interim plugging criteria (IPC) for fuel cycle 15. Because of the changes, text shift and repagenation are required. The change allows SG tubes with bobbin coil eddy current indications less than or equal to 2 volts at tube support plate intersections to remain in service, regardless of apparent depth of tube wall penetration, if as a result, the projected end-of-cycle (EOC) distribution of crack indications is shown to result in primary-to-secondary leakage less than 12.6 gpm in the faulted loop during a postulated steam line break event. Indications greater than 2 volts but less than or equal to 5.6 volts may remain in service if a motorized rotating pancake coil (MRPC) probe inspection does not detect degradation.

This amendment, specific to fuel cycle 15, would reduce the number of SG tubes plugged due to indications at support plate intersections. Reducing the number of plugged tubes provides ALARA benefits and maintains reactor coolant system (RCS) flow margin.

An assessment report addressing the effectiveness of the IPC methodology described in WCAP-13187, Revision 0, was completed following fuel cycle 13 and reported in submittal document AEP:NRC:1166J. The report concluded that the voltage distribution found by inspection at EOC 13 in 1994 is in good agreement with the projections made at EOC 12 in 1992. The voltage growth rates continue to be very small, with a maximum growth of 0.4 volts for fuel cycle 13 compared to 0.49 volts for fuel cycle 12. No tubes were found for which the bobbin coil voltage exceeded the 2 volt IPC repair limit at EOC 13. The maximum projected EOC 14 voltage based on EOC 13 voltage distribution is 2.0 volts using the NRC model and 1.9 volts using the industry model. Considering the results of this report, continuation of 2 volt IPC is justified for fuel cycle 15. Similar assessment and projection reports will be prepared at EOC 14 based on GL 94-XX reporting requirements.

II. APPLICATION OF DRAFT GL 94-XX REQUIREMENTS TO THE COOK NUCLEAR PLANT UNIT 1 SG IPC LICENSE AMENDMENT REQUEST FOR CYCLE 15

The Cook Nuclear Plant Unit 1, 2 volt IPC will be implemented per the guidance of GL 94-XX along with the latest industry data for burst and leakage data bases. NRC GL 94-XX will be factored into the Cook Nuclear Plant Unit 1 IPC as follows:

1) Analysts will be briefed regarding the possibility of primary water stress corrosion cracking (PWSCC) at tube support plate intersections. If PWSCC is found at the support plate intersections it will be reported to the NRC staff prior to startup.

2) The supporting data sets for calculation of burst probability and estimation of primary to secondary leakage during a postulated main steam line break will be those listed in Sections 2.a.1 and 2.b.3(1), respectively, of the GL.

3) Main steam line burst probability and leakage calculations will be performed following the guidance of GL 94-XX, Section 2, "Tube Integrity Evaluation." Calculations performed in support of the voltage-based repair criteria will follow the methodology described in WCAP-14277, "Steam Line Break Leak Rate and Tube Burst Probability Analysis Methods for Outside Diameter Stress Corrosion Cracking at Tube Support Plate Intersections," dated January 1995. The calculations will be performed prior to returning the SGs to service using the as-found EOC 14 voltage distribution. The projected EOC 15 voltage distribution results will be reported in the 90 day report. No distribution cutoff will be applied to the voltage measurement variability distribution for calculation of the projected EOC voltage distribution.

4) Inspection scope, data acquisition, and data analysis will be performed following the guidance of GL 94-XX, Section 3, "Inspection Criteria" and Appendix A, "NDE Data Acquisition and Analysis Guidelines" submitted by our letter AEP:NRC:1166H for the cycle 14 IPC. Motorized rotating pancake coil inspection will be done on all indications exceeding 1.5 volts. Motorized rotating pancake coil inspection will also be done on all intersections where copper signals, large mixed residuals, or dents larger than 5 volts interfere with detection of flaws.

   Probe wear inspections and re-inspections will be performed using the guidelines of Appendix A, Section A.2.3, as submitted by our letter AEP:NRC:1166H. If any of the last probe wear standard signal amplitudes prior to probe replacement exceed the +/- 15% limit, by a value of "X%", then any indications measured since the last acceptable probe wear measurement that are within "X%" of the plugging limit will be reinspected with the new probe. For example, if any of the last probe wear signal amplitudes prior to probe replacement were 17% above or below the initial amplitude, then the indications that are within 2% (17%-15%) of the plugging limit must be reinspected with the new probe. Alternatively, the voltage criterion may be lowered to compensate for the excess variation; for the case above, amplitudes ≥0.98 times the voltage criterion could be subject to repair.

Bobbin coil probe calibration will be performed using four 20% holes in the ASME calibration standard instead of the 100% through wall holes. This approach was concurred with by the NRC staff at the January 18, 1995, NRC/Industry meeting.

5) GL 94-XX, Section 5, "Operational Leakage Requirements," will be continued. The SG tube leakage limit of 150 gallons per day through each SG will be maintained as previously approved by the NRC for our last fuel cycle. Cook Nuclear Plant leakage monitoring methods provide timely leak detection, trending, and response to rapidly increasing leaks.

6) GL 94-XX, Section 6, "Reporting Requirements," will be implemented. It should be noted, as stated previously for Section 2, that the calculation of leakage and burst probability required prior to returning the SGs to service will be performed by use of the as-found EOC voltage distribution.

III. AEPSC COMMENTS/EXCEPTIONS TO GL 94-XX AND ASSOCIATED IMPACT TO AEPSC LICENSE AMENDMENT REQUEST FOR SG IPC FOR CYCLE 15

1) GL 94-XX, Section 1.b: Analyses performed by Westinghouse have shown that no tubes in the Cook Nuclear Plant Unit 1 SGs would be subject to collapse during a loss of coolant accident (LOCA) + safe shutdown earthquake (SSE) event. Therefore, no tubes are excluded based on this criteria. Items 1.b.2 and 1.b.3 are not applicable since these conditions do not exist in the Cook Nuclear Plant Unit 1 SGs. Series 51 SGs designed by Westinghouse do not have flow distribution baffle plates.

2) GL 94-XX, Section 3.b.2: Based on tube pull results from Cook Nuclear Plant Unit 1, copper deposits are not present on the tube outside diameter (OD) surfaces or in the tube support plate (TSP) crevice corrosion product. Similarly, Cook Nuclear Plant Unit 1 does not have evidence of "large mixed residual signals." Cook Nuclear Plant Unit 1 does not have a history of eddy current data which is difficult to interpret.

3) GL 94-XX, Section 3.c.2: Bobbin coil probes will continue to be calibrated against the 20% holes in the ASME calibration standard to remain consistent with the methodology used to develop the criteria.

4) GL 94-XX, Section 3.c.4: The requirement to reinspect all tubes prior to the last probe changeout if the wear measurement exceeds 15% is unnecessary. As acknowledged in the GL, a 5.6 volt repair criterion is justified, however, the repair criterion is currently limited to 2.0 volts. Such indications are expected to be well

within structural limits at EOC 15 conditions, particularly when Unit 1 growth rates are considered. Reinspection of indications necessitated by out of specification probe wear will be conducted according to Item 4 of page 2 of this attachment.

5) GL 94-XX, Section 4, addresses the need to perform tube pulls. Tube pulls cause significant outage extension, occupational radiation exposure, and significant direct cost. (As an example, removal of three tube samples during the upcoming Cook Nuclear Plant Unit 1 refueling outage is estimated to add two to three days to the outage critical path, have direct costs in the range of $0.8 - $1.3 million, and incur from 2-5 man rem exposure.) Therefore, tubes selected for pulling should be judiciously chosen and should justify the monetary expense and radiological exposure. Tubes should not be pulled merely to satisfy a chronological requirement. AEPSC believes that the imposition of this requirement during the next scheduled refueling outage will not enhance the current burst and leakage databases sufficiently to warrant the added cost. Justification for not performing a tube pull during the next outage at Cook Nuclear Plant Unit 1 is supplied below.

A) In 1992, nine Cook Nuclear Plant Unit 1 TSP intersections were removed for metallographic examination, burst testing and leak testing. Field bobbin coil voltages ranged from 1.0 to 2.0 volts, including four intersections reported as NDD. Burst pressures ranged from 9,100 to 11,200 psig for the reported indications and no intersections leaked during testing at 2560 psid. Examination of tubes after burst testing showed combinations of axially oriented intergranular stress corrosion cracking and intergranular cellular corrosion originating from the tube OD. Degradation morphology was dominated by the axially oriented degradation. In addition to the Unit 1 tube pulls supporting SG IPC, significant numbers of intersections were removed from Unit 2 in 1984 and 1986. In all cases, circumferentially oriented degradation was not detected.

B) Outside diameter stress corrosion cracking (ODSCC) degradation growth rate for Cook Nuclear Plant Unit 1 has decreased over the last two cycles and no intersections during the last outage had TSP intersection voltages exceeding 2.0 volts. This high level of performance is attributed to the unit operating at reduced temperature and pressure, improved secondary side chemistry control and sludge removal, and a conservative inspection/repair program. Based on the low growth rates at Unit 1 and chemistry control initiatives, EOC 14 voltages are expected to be well below 3.0 volts, and most likely less than 2.0 volts. Recent tube pulls at other plants

where the indication voltage was greater than 2.0 volts and where a larger number of beginning of cycle (BOC) indications were in the 2.0 volt range indicated no unexpected degradation morphology. The expected EOC 14 voltages at Unit 1 are well below the threshold for throughwall degradation of 2.8 volts (determined from a tube pull at another plant) and well below the threshold for SLB leakage of approximately 6.0 volts. During the current operating cycle, Unit 1 has not experienced any secondary side chemistry "excursions" which might support unexpected voltage growth or initiation of non-typical degradation morphologies.

C) The current requirement to perform an MRPC inspection of indications over 1.5 volts is sufficient to identify non-typical morphologies. For cases of significant cellular corrosion (identified at other plants), metallographic examination has shown that axially oriented degradation has dominated the morphology and burst test results. For such cases where significant circumferential components in a cellular morphology can influence burst, associated axial components would yield voltages far in excess of the 5 to 6 volt range, and such circumferentially oriented degradation would be of sufficient depth to be detected by the RPC probe.

IV. <u>10 CFR 50.92 EVALUATION</u>

BACKGROUND

Cook Nuclear Plant Unit 1 T/S Amendment 178 permitted the implementation of a 2.0 volt SG tube IPC for the 14th operating cycle of the Cook Nuclear Plant Unit 1 SGs. That license amendment, applicable only for the current operating cycle (cycle 14), required the repair of flaw-like bobbin indications above 2.0 volts. We are proposing use of a similar 2 volt interim repair criterion for the upcoming cycle 15.

The proposed IPC program for the Cook Nuclear Plant Unit 1 SGs follows the guidance and general intent of GL 94-XX to maintain tube structural and leakage integrity.

DESCRIPTION OF THE IPC REQUEST

As required by 10 CFR 50.91 (a)(1), an analysis is provided to demonstrate that the proposed license amendment to implement an IPC for the tube support plate elevation ODSCC occurring in the Cook Nuclear Plant Unit 1 SGs involves a no significant hazards consideration. The IPC utilizes

correlations between eddy current bobbin probe signal amplitude (voltage) and tube burst and leakage capability. The plugging criterion is based on testing of laboratory induced ODSCC specimens and on extensive examination of pulled tubes from operating SGs (industry wide - including three tubes pulled in 1992 representing nine intersections from Cook Nuclear Plant Unit 1 SGs.)

Consistent with GL 94-XX, the IPC program for Cook Nuclear Plant Unit 1 will include the following elements as listed under "1. Overview of the Voltage Repair Limit Approach", page 3 of GL 94-XX.

- *Perform an enhanced inspection of tubes, particularly at the TSP intersections.*

  A 100% bobbin coil inspection of hot leg tube support plate intersections and cold leg intersection down to the lowest cold leg support plate with known ODSCC indications will be performed. All flaw indications with bobbin voltages greater than 1.5 volts will be inspected by MRPC.

- *Utilize NDE data acquisition and analysis procedures that are consistent with the methodology used to develop the voltage-based repair limits.*

  The Cook Nuclear Plant Unit 1 IPC program will utilize procedures and techniques consistent with the methodologies used to establish the IPC as described in Section 3 of Enclosure 1 of the GL, with the exception that 20% throughwall holes will be used in the standard (Section 3.c.2 of GL 94-XX).

- *Repair or plug tubes that exceed the voltage limits.*

  Flaw-like signals adjacent to the TSP with bobbin voltages less than or equal to 2.0 volts will be allowed to remain in service. Flaw-like indications adjacent to the TSP with a bobbin voltage of greater than 2.0 volts but less than or equal to 5.6 volts may remain in service if MRPC inspection does not detect a flaw. Flaw indications with a voltage of greater than 5.6 volts will be repaired.

- *Determine the BOC voltage distribution.*

  Beginning of Cycle 15 voltage distribution will be established from the actual tube inspections to be performed during the next outage and will be established using current program methodology.

- *Project the EOC 15 distribution.*

An EOC voltage distribution will be established based on the EOC 14 ECT data. EOC 15 voltage distribution will be projected using Monte Carlo techniques as described in WCAP-14277 and will include allowance for eddy current uncertainty as defined in the GL and a conservative voltage growth rate allowance.

- *For the projected EOC voltage distribution, calculate leakage and conditional tube burst probability (and repair tubes if necessary).*

  Steam line break leakage will be calculated, as described in WCAP-14277, based on the EOC 15 projected voltage distribution. Projected leakage must remain below 12.6 gpm in the faulted loop for offsite dose estimates to remain within 10% of the 10 CFR 100 guidelines. This value was calculated, using Standard Review Plan methodology, prior to the Cycle 14 license amendment request and will not change for the upcoming cycle. Conditional tube burst probability will be calculated according to the methodology described in WCAP-14277. Consistent with the GL, if burst probability is found to be greater than 1×10^{-2}, the NRC will be consulted.

As prescribed in GL 94-XX, an evaluation of primary to secondary leakage (and subsequently offsite dose) is required for all plants implementing the IPC. All bobbin indications are included in the steam line break leakage analyses, along with consideration of the probability of detection. If the projected leakage exceeds 12.6 gpm in the faulted loop during a postulated steam line break event, the number of indications in which the IPC are applied is reduced through tube repair until the primary-to-secondary leakage limits are satisfied.

EVALUATION

Tube Degradation Characterization

In general, the degradation morphology occurring at the tube support plate intersections at plants in the U.S. can be described as axially oriented ODSCC. The degradation morphology at Cook Nuclear Plant Unit 1 is entirely compatible with the overall industry data base.

Steam Generator Tube Integrity

In the development of an IPC for Cook Nuclear Plant Unit 1, Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" and RG 1.83, "Inservice Inspection of PWR Steam Generator Tubes" are used as the bases for determining that SG tube integrity considerations are maintained within acceptable limits. Regulatory Guide 1.121 describes a method acceptable to the NRC staff for meeting General Design Criteria

(GDC) 14, 15, 31, and 32 by reducing the probability and consequences of SG tube rupture through determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service by plugging. This regulatory guide uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the ASME Code. For the tube support plate elevation degradation occurring in the Cook Nuclear Plant Unit 1 SGs, tube burst criteria are inherently satisfied during normal operating conditions by the presence of the tube support plate. The presence of the tube support plate enhances the integrity of the degraded tubes in that region by precluding tube deformation beyond the diameter of the drilled hole, thus precluding tube burst. Conservatively, no credit is taken in the development of the plugging criteria for the presence of the tube support plate during accident conditions. Based on the existing database for 7/8 inch tubing, burst testing shows that the safety requirements for tube burst margins during accident condition loadings can be satisfied with EOC bobbin coil signal amplitudes less than 9.6 volts, regardless of the depth of tube wall penetration of the cracking.

Upon implementation of these IPC, tube leakage considerations must also be addressed. It must be determined that the cracks will not leak excessively during all plant conditions. For the 2 volt interim tube plugging criteria developed for the Cook Nuclear Plant Unit 1 SG tubes, no leakage is anticipated during normal operating conditions even with the presence of potential throughwall cracks. The expected voltage which would support primary-to-secondary leakage at normal operating conditions is approximately 10 volts. No primary-to-secondary leakage at the TSP has been detected in U.S. plants. Relative to the expected leakage during accident condition loadings, the limiting event with respect to differential pressure experienced across the SG tubes is a postulated steam line break event. For 7/8 inch tubing, pulled tube data supports no leakage up to 2.81 volts and low probability of leakage between 2.81 and 6.0 volts, for both pulled tubes and model boiler specimens, at the bounding steam line break pressure differential of 2560 psi. Steam line break primary to secondary leakage will be calculated as prescribed in GL 94-XX and WCAP 14277, using EOC 14 eddy current data. This calculated leakage must be shown to be less than 12.6 gpm in the faulted loop.

Additional Considerations

The proposed amendment would preclude occupational radiation exposure that would otherwise be incurred by personnel involved in tube plugging or repair operations. By reducing non-essential tube plugging, the proposed amendment would minimize the loss of margin in the reactor coolant flow through the SG in LOCA analyses. The proposed amendment would avoid loss of margin in reactor coolant system flow and, therefore, assist in maintaining minimum flow rates in excess of that required for operation at

full power. Reduction in the amount of tube repair required can reduce the length of plant outages and reduce the time that the SG is open to the containment environment during an outage. The 100% eddy current bobbin probe inspection associated with implementation of the IPC will help to identify new areas of concern which may arise by providing a level of inservice inspection which is far in excess of the T/S requirements utilizing the 40% depth-based plugging limit for acceptable tube wall degradation.

SIGNIFICANT HAZARDS ANALYSIS

In accordance with the three factor test of 10 CFR 50.92(c), implementation of the proposed license amendment is analyzed using the following standards and found not to 1) 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; or 3) involve a significant reduction in margin of safety. Conformance of the proposed amendment to the standards for a determination of no significant hazards as defined in 10 CFR 50.92 (three factor test) is shown in the following paragraphs.

1) Operation of Cook Nuclear Plant Unit 1 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. Testing of model boiler specimens for free span tubing (no tube support plate restraint) at room temperature conditions show burst pressures in excess of 5000 psi for indications of outer diameter stress corrosion cracking with voltage measurements as high as 19 volts. Burst testing performed on pulled tubes from Cook Nuclear Plant Unit 1 with up to a 2.02 volt indication shows measured burst pressure in excess of 10,000 psi at room temperature. Burst testing performed on pulled tubes from other plants with up to 7.5 volt indications show burst pressures in excess of 6,300 psi at room temperatures. Correcting for the effects of temperature on material properties and minimum strength levels (as the burst testing was done at room temperature), tube burst capability significantly exceeds the safety factor requirements of RG 1.121. As stated earlier, tube burst criteria are inherently satisfied during normal operating conditions due to the proximity of the tube support plate. Test data indicates that tube burst cannot occur within the tube support plate, even for tubes which have 100% throughwall electric-discharge machined notches 0.75 inch long, provided the tube support plate is adjacent to the notched area. Since tube-to-tube support plate proximity precludes tube burst during normal operating conditions, use of the criteria must, therefore, retain tube integrity characteristics which maintain the RG 1.121 margin of safety of 1.43 times the bounding faulted condition (steam line break) pressure differential.

During a postulated main steam line break, the TSP has the potential to deflect during blowdown, thereby uncovering the intersection. Based on the existing data base, the RG 1.121 criterion requiring maintenance of a safety factor of 1.43 times the steam line break pressure differential on tube burst is satisfied by 7/8 inch diameter tubing with bobbin coil indications with signal amplitudes less than 9.6 volts, regardless of the indicated depth measurement. A 2.0 volt plugging criteria compares favorably with the 9.6 volt structural limit considering the previously calculated growth rates for ODSCC within Cook Nuclear Plant Unit 1 SGs. Considering a voltage growth component of 0.8 volts (40% voltage growth based on 2.0 volts BOC) and a nondestructive examination uncertainty of 0.40 volts (20% voltage uncertainty based on 2.0 volts BOC), when added to the BOC IPC of 2.0 volts, results in a bounding EOC voltage of approximately 3.2 volts for cycle 15 operation. A 6.4 volt safety margin exists (9.6 structural limit - 3.2 volt EOC = 6.4 volt margin).

For the voltage/burst correlation, the EOC structural limit is supported by a voltage of 9.6 volts. Using this structural limit of 9.6 volts, a BOC maximum allowable repair limit can be established using the guidance of RG 1.121. The BOC maximum allowable repair limit should not permit a significant number of EOC indications to exceed the 9.6 volt structural limit and should assure that acceptable tube burst probabilities are attained. By adding NDE uncertainty allowances and an allowance for crack growth to the repair limit, the structural limit can be validated. The previous IPC submittal established the conservative NDE uncertainty limit of 20% of the BOC repair limit. For consistency, a 40% voltage growth allowance to the BOC repair limit is also included. This allowance is extremely conservative for Cook Nuclear Plant Unit 1. Therefore, the maximum allowable BOC repair limit (RL) based on the structural limit of 9.6 volts can be represented by the expression:

$$RL + (0.2 \times RL) + (0.4 \times RL) = 9.6 \text{ volts, or,}$$

the maximum allowable BOC repair limit can be expressed as,

$$RL = 9.6 \text{ volt structural limit}/1.6 = 6.0 \text{ volts.}$$

This structural repair limit supports this application for cycle 15 IPC implementation to repair bobbin indications greater than 2.0 volts based on RPC confirmation of the indication. Conservatively, an upper limit of 5.6 volts will be used to repair bobbin indications which are above 2.0 volts but do not have confirming RPC calls.

The conservatism of this repair limit is shown by the EOC 13 (Spring 1994) eddy current data. The overall average voltage growth was determined to be only 1.4% (of the BOC voltage). In addition, the EOC 13 maximum observed voltage increase was 0.40 volts, and occurred in a tube with a BOC indication of 0.96 volts. The applicability of cycle 14 growth rates for cycle 15 operation will be confirmed prior to return to service of Cook Nuclear Plant Unit 1. Similar large structural margins are anticipated.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated main steam line break outside of containment but upstream of the main steam isolation valve represents the most limiting radiological condition relative to the IPC. In support of implementation of the IPC, it will be determined whether the distribution of crack indications at the tube support plate intersections at the end of cycle 15 are projected to be such that primary to secondary leakage would result in site boundary doses within a small fraction of the 10 CFR 100 guidelines. A separate calculation has determined this allowable steam line break leakage limit to be 12.6 gpm. Although not required by the Cook Nuclear Plant design basis, this calculation uses the recommended Iodine-131 transient spiking values consistent with NUREG-0800, and the T/S reactor coolant system activity limit of 1.0 micro curie per gram dose equivalent Iodine - 131. The projected steam line break leakage rate calculation methodology prescribed in GL 94-XX and WCAP 14277 will be used to calculate EOC 15 leakage, based on actual EOC 14 distributions and EOC 15 projected distributions. Due to the relatively low voltage growth rates at Cook Nuclear Plant Unit 1 and the relatively small number of indications affected by the IPC, steam line break leakage prediction per GL 94-XX is expected to be significantly less than the acceptance limit of 12.6 gpm in the faulted loop.

Prior to issue of GL 94-XX, projected EOC 14 leak rates were calculated, based on draft NUREG - 1477, for a total of twelve cases, the combination of six probability-of-leak correlations and two leak rate calculation methodologies. Results of the calculations show that the projected EOC 14 leak rates ranged from 0.001 gpm to 1.360 gpm. These results are well below the 12.6 gpm allowable; therefore, implementation of the 2 volt IPC during cycle 15 would not adversely affect SG tube integrity and results in acceptable dose consequences.

Current GL 94-XX methodology requires only the log-logistic probability of leakage correlation be used. Projected EOC 14 SLB leakage using this function was calculated to be only 0.001 gpm. Based on the relatively few numbers of intersections at Cook Nuclear Plant Unit 1 to which the IPC are applied and extremely small Cook

Nuclear Plant Unit 1 plant-specific growth rate, a similar value would be expected based on the EOC 14 eddy current data. The inclusion of all IPC intersections in the leakage model, along with application of a probability of detection of 0.6, will result in extremely conservative leakage estimations, especially so since close examination of the available data shows that indications of less than 2.8 volts will not be expected to leak during SLB conditions. All Unit 1 IPC indications are expected to be below 2.8 volts at the EOC 15 conditions.

The proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated within the Cook Nuclear Plant Unit 1 FSAR.

2) The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Implementation of the proposed SG tube IPC does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in an accident outside of the region of the tube support plate elevations. Neither a single or multiple tube rupture event would be expected in a SG in which the plugging criteria has been applied (during all plant conditions).

Specifically, we will continue to implement a maximum leakage rate limit of 150 gpd (0.1 gpm) per SG to help preclude the potential for excessive leakage during all plant conditions. The cycle 15 T/S limits imposed on primary to secondary leakage at operating conditions are: a maximum of 0.4 gpm (600 gpd) for all SGs with a maximum of 150 gpd allowed for any one SG.

The RG 1.121 criteria for establishing operational leakage rate limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture during faulted plant conditions. The 150 gpd limit should provide for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. Regulatory Guide 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before-break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded. The longest permissible crack is the length that provides a factor of safety of 1.43 against bursting at faulted conditions maximum pressure differential. A voltage amplitude of 9.6 volts for typical ODSCC corresponds to meeting this tube burst requirement at a lower 95% prediction limit on the burst

correlation coupled with 95/95 lower tolerance limit material properties. Alternate crack morphologies can correspond to 9.6 volts so that a unique crack length is not defined by the burst pressure versus voltage correlation. Consequently, typical burst pressure versus through-wall crack length correlations are used below to define the "longest permissible crack" for evaluating operating leakage limits.

Consistent with the Cycle 13 and Cycle 14 license amendment requests for IPC and Section 5 of Enclosure 1 of the GL, operational leakage limits will remain at 150 gpd per SG. Axial cracks leaking at this level are expected to provide leak before break (LBB) protection at both the SLB pressure differential of 2560 psi and, while not part of any established LBB methodology, LBB protection will also be provided at a value of 1.43 times the SLB pressure differential. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for steam line break conditions. Additionally, this leak-before-break evaluation assumes that the entire crevice area is uncovered during blowdown. Partial uncovery will provide benefit to the burst capacity of the intersection.

3) The proposed license amendment does not involve a significant reduction in margin of safety.

The use of the voltage based bobbin probe interim tube support plate elevation plugging criteria at Cook Nuclear Plant Unit 1 is demonstrated to maintain SG tube integrity commensurate with the criteria of RG 1.121. Regulatory Guide 1.121 describes a method acceptable to the NRC staff for meeting GDC 14, 15, 31, and 32 by reducing the probability or the consequences of SG tube rupture. This is accomplished by determining the limiting conditions of degradation of SG tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criteria, even under the worst case conditions, the occurrence of ODSCC at the tube support plate elevations is not expected to lead to a SG tube rupture event during normal or faulted plant conditions. The EOC 15 distribution of crack indications at the tube support plate elevations will be confirmed by analysis and calculation to result in acceptable primary to secondary leakage during all plant conditions and that radiological consequences are not adversely impacted.

In addressing the combined effects of a LOCA and SSE on the SG component (as required by GDC 2), it has been determined that tube collapse may occur in the SGs at some plants. This is the case as the tube support plates may become deformed as a result of lateral

loads at the wedge supports at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with SG tube collapse. First, the collapse of SG tubing reduces the RCS flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase peak clad temperature. Second, there is a potential that partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse.

Consequently, since the leak-before-break methodology is applicable to the Cook Nuclear Plant Unit 1 reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. The limiting LOCA event becomes either the accumulator line break or the pressurizer surge line break. Loss of coolant accident loads for the primary pipe breaks were used to bound the Cook Nuclear Plant Unit 1 smaller breaks. The results of the analysis using the larger break inputs show that the LOCA loads were found to be of insufficient magnitude to result in SG tube collapse or significant deformation.

Addressing RG 1.83 considerations, implementation of the bobbin probe voltage based interim tube plugging criteria of 2.0 volts is supplemented by enhanced eddy current inspection guidelines to provide consistency in voltage normalization, a 100% eddy current inspection sample size at the tube support plate elevations per T/S, and MRPC inspection requirements for the larger indications left in service to characterize the principal degradation as ODSCC.

As noted previously, implementation of the tube support plate elevation plugging criteria will decrease the number of tubes which must be repaired. The installation of SG tube plugs reduces the RCS flow margin. Thus, implementation of the IPC will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to plant safety as defined in the Final Safety Analysis Report or any Bases of the plant T/Ss.