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SUBJECT: Application for amend to License DPR-58,incorporating 2.0 volt steam generator tube support plate interim plugging criteria for Cycle 14.

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AEP:NRC:1166L

Donald C. Cook Nuclear Plant Unit 1
Docket No. 50-315
License No. DPR-58
TECHNICAL SPECIFICATION CHANGES TO
INCORPORATE 2.0 VOLT INTERIM STEAM
GENERATOR TUBE SUPPORT PLATE PLUGGING
CRITERION FOR FUEL CYCLE 14

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

Attn: T. E. Murley

February 15, 1994

Dear Dr. Murley:

This letter and its attachments constitute an application for an exigent amendment to the Technical Specifications (T/Ss) for the Donald C. Cook Nuclear Plant Unit 1. Specifically, we are requesting to incorporate a 2.0 volt steam generator tube support plate interim plugging criterion for Fuel Cycle 14. As discussed with your staff on February 9, 1994, the reason for making the request on an exigent basis is that the change is associated with steam generator repairs during the current Unit 1 refueling outage. The repairs are currently scheduled to begin March 6, 1994. Therefore, we request approval of this amendment request by March 4 in order to avoid outage delays.

By letter dated December 15, 1993 (AEP:NRC:1166H), we requested similar steam generator tube interim plugging criteria T/S changes. This letter does not supersede the previous submittal. The changes differ from our previous submittal in that this submittal incorporates a 2.0 volt criterion versus the previous 1.0 volt limit. The upper voltage limit for disposition of indications by rotating pancake coil inspection is also changed from 3.1 volts to 3.6 volts. Additionally, predicted end-of-cycle leakage in the faulted loop for a postulated steamline break is limited to 12.6 gpm, versus 1.0 gpm. This is based on a radiological analysis performed in accordance with the Standard Review Plan, including iodine spiking and T/S reactor coolant system activity limits. The T/S changes requested in this submittal are similar to a December 9, 1993, request made by the Farley Nuclear Plant Unit 1.

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As discussed with your staff on February 9, a rotating pancake coil inspection will be performed for all tubes with an indication above 1.0 volt, despite the change in the interim plugging criteria to 2.0 volts. It is anticipated that the standard bobbin probe (0.720 inches) will be the probe used for tube plugging determination in conjunction with the interim plugging criteria. If a smaller bobbin probe is used, it is our understanding that its use must be supported by a rigorous statistical analysis. If the smaller bobbin probe is to be used, the details of the statistical analysis will be the subject of separate correspondence.

As discussed in our submittal AEP:NRC:1166H, we stated our position regarding not removing tubes during the 1994 refueling outage. Changing the interim plugging criteria from 1.0 to 2.0 volts does not change the basis for our position. Also, that submittal provided an appendix entitled "NDE Data Acquisition and Analysis Guidelines." Although this appendix was written specific to an IPC of 1.0 volts, it is still applicable in the general sense with an IPC of 2.0 volts, and therefore, per discussions with your staff of February 9, 1994, will not be revised for this letter.

Attachment 1 contains a description of the proposed T/S changes as well as the 10 CFR 50.92 no significant hazards evaluation. Attachment 2 contains the existing T/S pages marked to reflect the changes. Attachment 3 contains the proposed revised T/S pages. Attachment 4 contains the radiological analysis that supports the 12.6 gpm end-of-cycle leakage limit discussed above.

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 by the Nuclear Safety and Design Review Committee.

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.

This letter is submitted pursuant to 10 CFR 50.30(b), and, as such, an oath statement is attached.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President

dr

Attachments

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

STATE OF OHIO)
COUNTY OF FRANKLIN)

E. E. Fitzpatrick, being duly sworn, deposes and says that he is the Vice President of licensee Indiana Michigan Power Company, that he has read the forgoing TECHNICAL SPECIFICATION CHANGES TO INCORPORATE 2.0 VOLT INTERIM STEAM GENERATOR TUBE SUPPORT PLATE PLUGGING CRITERION FOR FUEL CYCLE 14 and knows the contents thereof; and that said contents are true to the best of his knowledge and belief.

E E Fitzpatrick

Subscribed and sworn to before me this 15th

day of February, 19 94.

Rita D. Hill
NOTARY PUBLIC

RITA D. HILL
NOTARY PUBLIC, STATE OF OHIO
MY COMMISSION EXPIRES 6-28-94

ATTACHMENT 1 TO AEP:NRC:1166L

DESCRIPTION AND JUSTIFICATION OF CHANGES

10 CFR 50.92 ANALYSIS FOR CHANGES
TO THE DONALD C. COOK NUCLEAR PLANT
UNIT 1 TECHNICAL SPECIFICATIONS

I. INTRODUCTION

This amendment request proposes a change to T/S 4.4.5 (Steam Generators) to incorporate a revised criterion for steam generator tube support plate interim plugging criterion (IPC). The changes allow steam generator tubes with indications less than or equal to 2.0 volts to remain in service, regardless of depth of tube wall penetration, if as a result, the projected end-of-cycle (EOC) distribution of crack indications is verified to result in primary-to-secondary leakage less than 12.6 gpm in the faulted loop during a postulated steamline break event. Indications greater than 2.0 volts but less than or equal to 3.6 volts can remain in service if a rotating pancake coil probe (RPC) inspection does not detect degradation. The amendment request is specific to Fuel Cycle 14 only. The purpose of the amendment is to reduce the number of tubes required to be plugged. This has benefits both from an ALARA perspective as well as the perspective of maximizing RCS flow margin.

A 1.0 volt IPC was granted for Cook Nuclear Plant Unit 1, via T/S Amendment 166, for the previous fuel cycle (Cycle 13).

By letter dated December 15, 1993 (AEP:NRC:1166H), we requested similar steam generator tube interim plugging criteria T/S changes for Cycle 14. This letter does not supersede the previous submittal. The changes differ from our previous submittal in that this submittal incorporates a 2.0 volt criteria versus the previous 1.0 volt limit. The upper voltage limit for RPC confirmation is changed from 3.1 volts to 3.6 volts. Additionally, predicted end-of-cycle leakage in the faulted loop for a postulated steamline break is limited to 12.6 gpm, versus 1.0 gpm. This is based on a radiological analysis performed in accordance with the Standard Review Plan, including iodine spiking and T/S limits on reactor coolant system activity. (The previous submittal did not include the effects of iodine spiking on the dose analysis and assumed reactor coolant activity corresponding to 1% failed fuel.) The T/S changes requested in this submittal are similar to a December 9, 1993, request made by the Farley Nuclear Plant Unit 1.

As discussed with your staff on February 9, an RPC inspection will be performed for all tubes with an indication above 1.0 volt, despite the change in the interim plugging criteria from 1.0 volt to 2.0 volts. It is anticipated that the standard bobbin probe (0.720 inches) will be the probe used for tube plugging determination in conjunction with the interim plugging criteria. If a smaller bobbin probe is used, it is our understanding that its use must be supported by a rigorous statistical analysis. If the smaller bobbin probe is to be used, the details of the statistical analysis will be the subject of separate correspondence.

Our 1.0 volt IPC submittal (AEP:NRC:1166H) provided an appendix entitled "NDE Data Acquisition and Analysis Guidelines." Although this appendix was written specific to an IPC of 1.0 volt, it is still applicable in the general sense with an IPC of 2.0 volts, and therefore, per discussions with your staff of February 9, 1994, will not be revised for this submittal.

Assessment of IPC Methodology

An assessment of the methodology described in WCAP-13187, Revision 0 will be conducted for the IPC. It will address discrepancies between predicted and actual EOC voltage distributions. The assessment will include:

a. EOC 12 voltage distribution - indications found during the inspection regardless of RPC verification results.

b. Cycle 12 growth rate (i.e. from beginning of cycle (BOC) 12 to EOC 12).

c. EOC 12 repaired indications voltage distribution - distribution of indications presented in (a) above that were repaired (if plugged or sleeved).

d. Voltage distribution for indications left in service at the BOC 13 regardless of RPC confirmation - obtained from (a) and (c) above.

e. Voltage distribution for indications left in service at the BOC 13 that were confirmed by RPC to be crack-like or not RPC inspected.

f. Non-destructive examination uncertainty distribution used in predicting the EOC 13 voltage distribution.

g. Projected EOC 13 voltage distribution using the methodology in WCAP-13187, Revision 0.

h. Actual EOC 13 voltage distribution - indications found during the inspection regardless of RPC confirmation.

i. Cycle 13 growth rate (i.e. from BOC 13 to EOC 13).

j. EOC 13 repaired indications voltage distribution - distribution of indications presented in (h) above that were repaired (i.e. plugged or sleeved).

k. Voltage distribution for indications left in service at the BOC 14 regardless of RPC confirmation - obtained from (h) and (j) above.

l. Voltage distribution for indications left in service at the BOC 14 that were confirmed by RPC to be crack-like or not RPC inspected.

m. Nondestructive examination uncertainty distribution used in predicting the EOC 14 voltage distribution.

n. Projected EOC 14 voltage distribution using the methodology in WCAP-13187, Revision 0.

The total assessment, (a) through (n) will be submitted approximately 10 weeks from completion of steam generator inspections. However, per T/S requirements the steamline break leakage analysis performed under T/S 4.4.5.4.a.10 will be reported to the NRC Staff prior to restart for Fuel Cycle 14.

II. DESCRIPTION OF THE CHANGES

The proposed changes are those necessary to incorporate a 2.0 volt IPC for Cook Nuclear Plant Unit 1 Fuel Cycle 14. The specific changes are as follows:

1. T/S 4.4.5.2.d, 4.4.5.5.e, 3.4.6.2.c, Bases 3/4.4.5, Bases 3/4.4.6.2

Reference to Fuel Cycle 13 for the IPC is changed to Fuel Cycle 14.

2. T/S 4.4.5.4.a.10

The definition of IPC is modified such that steam generator tubes with indications less than or equal to 2.0 volts (vs. 1.0 volt) can remain in service, regardless of depth of tube wall penetration, if as a result, the projected end-of-cycle distribution of crack indications is verified to result in primary-to-secondary leakage less than 12.6 gpm (vs. 1.0 gpm) in the faulted loop during a postulated steamline break event. Indications greater than 2.0 volts but less than or equal to 3.6 volts (vs. 1.0 volt and 4.0 volts) can remain in service if a RPC inspection does not detect degradation.

3. Bases 3/4.4.6.2

The end-of-cycle primary to secondary leakage must be less than 12.6 gpm in the faulted loop, versus the present 1.0 gpm, for a postulated steamline break event. The leakage assumed in the accident analysis is changed from 120 gpm to 12.6 gpm (for the Cycle 14 IPC only).

III. 10 CFR 50.92 EVALUATION

BACKGROUND

Cook Nuclear Plant Unit 1 T/S Amendment 166 permitted the implementation of a 1.0 volt steam generator tube interim plugging criteria (IPC) for the 13th operating cycle of the Cook Nuclear Plant Unit 1 steam generators. This license amendment was applicable only for the previous cycle (Cycle 13), and required the repair of flaw-like bobbin indications above 1.0 volt. Based upon tube pull results from Cook Nuclear Plant Unit 1 and other plants gathered since the start of Cycle 13, we are proposing use of a 2.0 volt interim repair criterion for the upcoming Cycle 14.

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 interim tube plugging criteria for the tube support plate elevation outer diameter stress corrosion cracking (ODSCC) occurring in the Cook Nuclear Plant Unit 1 steam generators 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 criteria is based on testing of laboratory induced ODSCC specimens, and extensive examination of pulled tubes from operating steam generators (industry wide - including 3 tubes representing 6 intersections from the Cook Nuclear Plant Unit 1 steam generators.)

The interim plugging criteria can be described by the following elements:

1. A 100% bobbin coil inspection of hot leg tube support plate intersections and cold leg intersections down to the lowest cold leg tube support plate with known ODSCC indications will be performed.

2. Flaw-like signals adjacent to the tube support plates with bobbin voltages less than or equal to 2.0 volts will be allowed to remain in service.

3. Flaw-like signals adjacent to the tube support plate with a bobbin voltage of greater than 2.0 volts will be repaired except as noted in Item 4.

4. Flaw-like signals adjacent to the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to 3.6 volts may remain in service if RPC inspection does not detect a flaw. Flaw indications with a bobbin voltage greater than 3.6 volts will be repaired.

5. As part of a sample inspection program to help ensure that additional degradation modes are not occurring, all flaw indications with bobbin voltages greater than 1.0 volt but less than or equal to 2.0 volts will be inspected by RPC.

6. An end-of-cycle voltage distribution will be established based upon the end-of-Cycle 13 eddy current data. Based upon this distribution, postulated steamline break leakage will be estimated based on the guidance of draft NUREG 1477. Projected leakage must remain below 12.6 gpm in the faulted loop in order for offsite dose estimates to remain within 10% of the 10 CFR 100 guidelines.

As prescribed in draft NUREG-1477, an evaluation of primary to secondary leakage (and subsequently offsite dose) is required for all plants implementing the interim plugging criteria (IPC). Per draft NUREG-1477, all bobbin indications are included in the steamline break leakage analyses along with the consideration of probability of detection (POD). If the projected leakage exceeds 12.6 gpm in the faulted loop during a postulated steamline break event, the number of indications in which the interim plugging criteria 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 interim plugging criteria 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 steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC staff for meeting General Design Criteria 14, 15, 31, and 32 by reducing the probability and consequences of steam generator 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 steam generators, 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 end of cycle bobbin coil signal amplitudes less than 9.6 volts, regardless of the depth of tube wall penetration of the cracking.

Upon implementation of the plugging criteria, tube leakage considerations must also be addressed. It must be determined that the cracks will not leak excessively during all plant conditions. For the 2.0 volt interim tube plugging criteria developed for the Cook Nuclear Plant Unit 1 steam generator tubes, no leakage is anticipated during normal operating conditions even with the presence of potentially throughwall cracks. No leakage during normal operating conditions has been observed in the field for crack indications with signal amplitudes up to 7.7 volts (3/4 inch tubes). Voltage correlation to 7/8 inch tubing size would result in an expected voltage of about 10 volts. No primary to secondary leakage at the tube support plates (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 steamline 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 steamline break pressure differential of 2560 psi. Steamline break primary to secondary leakage will be calculated as prescribed in Section 3.3 of draft NUREG-1477 (using a primary-to-secondary pressure differential of 2560 psid) once EOC 13 eddy current data is reduced. 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 steam generator in LOCA analyses. The proposed amendment would avoid loss of margin in reactor coolant system flow and, therefore, assist in demonstrating that minimum flow rates are maintained 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 steam generator 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:

1) Operation of the Donald C. 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 (EDM) notches 0.75 inch long, provided that 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 retain tube integrity characteristics which maintains the R.G. 1.121 margin of safety of 1.43 times the bounding faulted condition (steamline break) pressure differential.

During a postulated main steamline 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 steamline 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 the Cook Nuclear Plant Unit 1 steam generators. Considering a voltage growth component of 0.8 volts (40% voltage growth based on 2.0 volts BOC) and an NDE uncertainty of 0.40 volts (20% voltage uncertainty based on 2.0 volts BOC), when added to the BOC interim plugging criteria of 2.0 volts results in a bounding EOC voltage of approximately 3.2 volts for Cycle 14 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 beginning of cycle (BOC) maximum allowable repair limit can be established using the guidance of RG 1.121. The BOC maximum allowable repair limit should not permit the existence of EOC indications which exceed the 9.6 volt structural limit. By adding NDE uncertainty allowances and an allowance for crack growth to the repair limit, the structural limit can be validated. Previous IPC submittals have 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 (RL) 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 14 IPC implementation to repair bobbin indications greater than 2.0 volts independent of RPC confirmation of the indication. Conservatively, an upper limit of 3.6 volts will be used to assess tube integrity for those 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 12 (Summer 1992) eddy current data. The overall average voltage growth was determined to be only 2.2% (of the BOC voltage), with a 12% average voltage growth for indications less than 1.0 volt BOC and a 1% average voltage growth for indications greater than 0.75 volts at the BOC. In addition, the EOC 12 maximum observed voltage increase was found to be 0.49 volts, and occurred in a tube initially less than 1.0 volt BOC. The applicability of Cycle 13 growth rates for Cycle 14 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 steamline 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 interim plugging criteria, it will be determined whether the distribution of crack indications at the tube support plate intersections at the end of Cycle 14 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 steamline 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 steamline break leakage rate calculation methodology prescribed in Section 3.3 of draft NUREG-1477 will be used to calculate EOC 14 leakage. 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, steamline break leakage prediction per draft NUREG-1477 is expected to be less than the acceptance limit of 12.6 gpm in the faulted loop.

Application of the criteria requires the projection of postulated steamline break leakage, based on the EOC voltage distribution. EOC voltage distribution is developed using EOC-13 eddy current results and a voltage measurement uncertainty. The data indicates that a threshold voltage of 2.81 volts would result in throughwall cracks long enough to leak at steamline break conditions. Draft NUREG-1477 requires that all indications to which the IPC are applied must be included in the leakage projection. Tube pull results from Cook Nuclear Plant Unit 1 indicate that tube wall degradation of greater than 40% throughwall was detectable either by the bobbin or RPC probe. The tube with maximum throughwall penetration of 56% (42% average) had a voltage of 2.02 volts. This indication also was the largest recorded bobbin voltage from the EOC 12 leakage of 2.81 volts, inclusion of all IPC intersections in the leakage model is quite conservative. Therefore, as implementation of the 2.0 volt interim plugging criteria during Cycle 14 does not adversely affect steam generator tube integrity and results in acceptable dose consequences 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 steam generator tube interim plugging criteria 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; no ODSCC is occurring outside the thickness of the tube support plates. Neither a single or multiple tube rupture event would be expected in a steam generator 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 steam generator to help preclude the potential for excessive leakage during all plant conditions. The Cycle 14 T/S limits on primary to secondary leakage at operating conditions is a maximum of 0.4 gpm (600 gpd) for all steam generators, or, a maximum of 150 gpd for any one steam generator.

The RG 1.121 criterion 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. RG 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 (LTL) 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.

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.

The single through-wall crack lengths that result in tube burst at 1.43 times the steamline break pressure differential (1.43 x 2560 psi = 3660 psi) and the steamline break pressure differential alone (2560 psi) are approximately 0.53 inch and 0.84 inch, respectively. A leak rate of 150 gpd will provide for detection of 0.42 inch long cracks at nominal leak rates and 0.61 inch long cracks at the lower 95% confidence level leak rates. Since tube burst is precluded during normal operation due to the proximity of the TSP to the tube and the potential exists for the crevice to become uncovered during steamline break conditions, the leakage from the maximum permissible crack must preclude tube burst at steamline break conditions. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for steamline 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 steam generator tube integrity commensurate with the criteria of RG 1.121. RG 1.121 describes a method acceptable to the NRC staff for meeting GDCs 14, 15, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator 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 steam generator tube rupture event during normal or faulted plant conditions. The EOC 14 distribution of crack indications at the tube support plate elevations will be confirmed 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 loss of coolant accident (LOCA) + safe shutdown earthquake (SSE) on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators 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 steam generator tube collapse. First, the collapse of steam generator 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 (PCT). 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. LOCA 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 steam generator 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 RPC 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 steam generator tube plugs reduces the RCS flow margin. Thus, implementation of the interim plugging criteria 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.