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SUBJECT: Application for amend to License DPR-58, changing TS to allow interim plugging criteria of 1.0 volt for Cycle 14. Technical summary supporting validity & continued use of Westinghouse Rept WCAP-13187 encl.

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AEP:NRC:1166G

Donald C. Cook Nuclear Plant Unit 1
Docket No. 50-315
License No. DPR-58
TECHNICAL SPECIFICATIONS CHANGE TO ALLOW
INTERIM PLUGGING CRITERIA OF 1.0 VOLT FOR
CYCLE 14

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

Attn: T. E. Murley

March 10, 1993

Dear Dr. Murley:

This letter and its attachments constitute an application for an amendment to the Technical Specifications (T/Ss) of Donald C. Cook Nuclear Plant Unit 1 in accordance with 10 CFR 50.90. It is requested that the amendment to the T/Ss for steam generator interim plugging criteria, as outlined in the NRC letter of July 29, 1992 for license Amendment No. 166 for fuel cycle 13, be extended to fuel cycle 14 of the Cook Nuclear Plant Unit 1. This extension is essential while our T/S ammendment to allow alternate plugging criteria is being reviewed by the NRC (AEP:NRC:1166, March 20, 1992).

Attachment 1 provides a technical summary of the recent steam generator tube interim plugging criteria voltage evaluations and tube sample analysis results. These evaluations and results support the validity and continued use of the Westinghouse Report WCAP-13187 in support of the 1 volt interim plugging criteria for fuel cycle 14. No technical changes were made to the interim plugging criteria T/Ss requirements used for fuel cycle 13. T/S changes were made only to reference fuel cycle 14 in the appropriate T/S paragraphs where fuel cycle 13 was previously referenced.

A detailed description of the proposed changes and our analyses concerning significant hazards considerations are repeated from our previous submittal and are included in Attachment 2 to this letter for your convenience. Attachment 3 contains the proposed

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revised T/S pages. Attachment 4 contains the marked-up copies of the existing T/Ss.

We believe that the proposed changes will not result in (1) a significant change in the types of effluents or a significant increase in the amount of any effluents 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 the corporate Nuclear Safety and Design Review Committee.

In compliance with the requirements of 10CFR50.91(b)(1), copies of this letter and its attachments have been transmitted to Mr. J. R. Padgett of the Michigan Public Service Commission and to the Michigan Department of Public Health.

Please contact us if you have any questions concerning this license amendment request.

This letter is submitted pursuant to 10 CFR 50.54(f) and, as such, an oath statement is enclosed.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President

eh

Attachments

Enclosure

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

bc: S. J. Brewer
D. H. Malin/K. J. Toth
J. D. Benes/J. R. Jensen
M. L. Horvath - Bridgman
J. B. Shinnock
W. G. Smith, Jr.
W. M. Dean, NRC - Washington, D. C.
AEP:NRC:1166G
DC-N-6015.1

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 response to AEP:NRC:1166G - Technical Specification Change To Allow Interim Plugging Criteria Of 1.0 Volt For 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 12th

day of March, 1993.

Rita D. Hill
NOTARY PUBLIC

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

ATTACHMENT 1 TO AEP:NRC:1166G

TECHNICAL SUMMARY OF THE RECENT

STEAM GENERATOR TUBE INTERIM PLUGGING CRITERIA

VOLTAGE EVALUATIONS

AND

TUBE SAMPLE ANALYSIS RESULTS

The technical basis for implementing the interim plugging criteria for fuel cycle 13 was based on submittal documents AEP:NRC:1166, 1166A, 1166B, 1166C, and 1166D. Included as an attachment was Westinghouse Report, WCAP-13187 titled "D.C. Cook Unit 1 Steam Generator Tube Plugging Criteria for Indications at Tube Support Plates" dated March 1992. The WCAP contains the technical approach for achieving assurance of tube structural integrity based on eddy current bobbin coil voltage repair criteria. Application of the WCAP tube repair voltage criteria is based on voltage growth rates from the prior two operating cycles of 0.8 and 0.7 volts. Analysis of voltage growth rate done for fuel cycle 13, at the end of fuel cycle 12 in 1992, showed the largest voltage growth was 0.49 volts. This is well below the projected growth rates referenced by the WCAP. Analysis for the maximum end of cycle voltage projected for indications left in service below 1.0 volt was 1.40 volts. This indication voltage is well within the structural limits proposed by the WCAP structural voltage limits for potential tube leakage and burst. NRC submittal letter AEP:NRC:1166F contains the full report on evaluations performed to estimate the projected EOC voltage growth rate as well as leakrate and burst probability.

In addition, tube samples were removed from steam generator No. 12 as part of the T/Ss alternate plugging criteria for metallographic evaluation, leak testing, and burst testing. The following table summarizes the results on those tube samples leak and burst tested at the first and second support plate (SP) elevations.

Summary Table
Tube Leak and Burst Test Data

Location Tube/SP	Field E/C Call	Leak Test, Psig	Burst Test, Psig	SEM Fractography Max Depth (%)	SEM Fractography Ave Depth (%)
R11-C60/SPI	1.1 Volt, BC SAI, RPC	Not Done	9,100	54	32
R11-C60/SP2	1.4 Volt, BC MAI, RPC	Not Done	9,350	52	38
R18-C16/SP1	1.4 Volt, BC SAI, RPC	>2,650 (ET)* No Leakage	10,725	48	28
R18-C21/SP1	2.0 Volt, BC SAI, RPC	>2,650 (ET)* No Leakage	10,200	56	42
R18-C21/SP2	NDD, BC NDD, RPC	Not Done	11,200	38	No Data

*ET=Test done at elevated temperature.

The corrosion was confined to the support plate region and had combinations of axially oriented intergranular stress corrosion cracking and intergranular cellular corrosion. The corrosion was of OD origin, and it occurred in either 360° bands or isolated patches.

The SP2 region of tube R18-C21 had the shallowest maximum crack depth (38% throughwall) found. Its eddy current signal would not be expected to be detected with a high probability.

The first support plate crevice regions of tubes R18-C16 and R18-C21 were leak tested at elevated temperature and pressure. Neither tube leaked through existing corrosion networks at normal operating conditions (1500 psi differential pressure with primary side at 2250 psi and secondary side at 750 psi) or at steam line break conditions (2650 psi differential pressure with the primary side at 3000 psi and the secondary side at 350 psi).

Room temperature burst tests were conducted at a higher pressurization rate of 1000 psi/sec on SP1 of tubes R11-C60, R18-C16 and R18-C21 and SP2 of tubes R11-C60 and R18-C21. All burst at high burst pressures (9,100; 10,725; 10,200; 9,350 and 11,200 psig, respectively) and had axial burst openings. Virgin tubing burst typically between 11,500 and 12,000 psig.

In conclusion, the results of the voltage growth rate evaluations and tube sample leak and burst were well within the guidelines established in the WCAP. Also, the voltage growth rate is almost half of what was found for the two prior fuel cycles. Therefore, based on a low voltage growth rate, and leak and burst test results, the WCAP is still considered a technically appropriate and bounding document applicable to the next fuel cycle, cycle 14.

Furthermore, the eddy current inspection, reporting, and leakage requirements as previously stated in the T/Ss for fuel cycle 13, will be maintained for fuel cycle 14.

ATTACHMENT 2 to AEP:NRC:1166G

NO SIGNIFICANT HAZARDS CONSIDERATION

EVALUATION IN SUPPORT OF THE

INTERIM PLUGGING CRITERIA

INTRODUCTION

A license amendment is proposed to preclude unnecessarily plugging steam generator tubes due to the occurrence of outer diameter initiated stress corrosion cracking (ODSCC) at the tube support plates in the Cook Nuclear Plant Unit 1 steam generators. Using the existing Technical Specifications (T/Ss) steam generator tube plugging criteria of 40% tube wall penetration as determined by non-destructive examination (NDE), many of the tubes with crack indications would needlessly have to be removed from service. The interim plugging criteria for tube support plate elevation ODSCC occurring in the Cook Nuclear Plant Unit 1 steam generators may result in tubes with both partial and through-wall cracks returning to service. In the limiting case, it is demonstrated that the presence of through-wall cracks alone is not reason enough to remove a tube from service.

DESCRIPTION OF THE AMENDMENT 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 steam generator tube plugging criteria for the tube support plate elevations at Cook Nuclear Plant Unit 1 involves no significant hazards considerations. The interim plugging criteria involve a correlation between eddy current bobbin coil signal amplitude (voltage) and tube burst and leakage capability. The plugging criteria is based on testing of laboratory-induced ODSCC specimens, extensive examination of pulled tubes from operating steam generators (industry wide), and field experience with leakage due to indications at the tube support plates (world wide).

Specifically, crack indications with bobbin coil voltages less than or equal to 1.0 volt, regardless of indicated depth, do not require remedial action if postulated steam line break leakage can be shown to be acceptable. Crack indications with bobbin coil signal amplitudes exceeding 1.0 volt must be either plugged or repaired.

The proposed amendment would maintain the previously modified T/Ss 3.4.5 "Steam Generators," 3.4.6, "Reactor Coolant System Leakage," and the associated bases incorporated for the 1 volt interim plugging criteria. These provide tube inspection requirements and acceptance criteria to determine the level of degradation for which a tube experiencing ODSCC at the tube support plate elevations may remain in service in the Cook Nuclear Plant Unit 1 steam generators.

EVALUATION

Steam Generator Tube Integrity Discussion

In the development of the interim plugging criteria, Regulatory Guides (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," and 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. Regulatory Guide 1.121 describes a method acceptable to the NRC staff for meeting General Design Criteria (GDC) 2, 4, 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. Tubes with unacceptable cracking, as established by inservice inspection, should be repaired or removed from service by plugging. This RG 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. It is not certain whether the tube support plate would function to provide a similar constraining effect during accident condition loadings. Therefore, no credit is taken in the development of the plugging criteria for the presence of the tube support plate during accident condition loadings. Conservatively, based on the existing data base, burst testing shows that the safety requirements for tube burst margins during both normal and accident condition loadings can be satisfied with bobbin coil signal amplitudes less than 6.8 volts, regardless of the depth of tube wall penetration cracking. Regulatory Guide 1.83 describes a method acceptable to the NRC staff for implementing GDC 14, 15, 31, and 32 through periodic inservice inspection for the detection of significant tube wall degradation.

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 interim tube plugging criteria developed for the Cook Nuclear Plant Unit 1 steam generator tubes, little or no leakage is expected during normal operating conditions even with the presence of through-wall cracks. Industry wide, the crack morphology of SCC at tube support plate intersections is best described as short, tight, axially oriented microcracks separated by ligaments of non-degraded material. The same morphology is judged to be present in the Cook Nuclear Plant Unit 1 steam generators. Tube pull examination results from 1983 indicated axial SCC in its early stages. The depths of the evidenced degradation (determined by destructive exam) showed the maximum depth of penetration to be approximately 10% through-wall. The rotating pancake coil (RPC) testing performed during the 1989 and 1992 outages has confirmed that axially oriented SCC cracks exist at the tube support plate intersections. Based on the RPC testing results and relatively small amplitude bobbin voltages coupled with low bobbin voltage growth rates, it is concluded that axially oriented ODSCC best defines the degradation morphology occurring at the Cook Nuclear Plant Unit 1 tube support plate intersections. Tube pulls during the summer 1992 refueling outage were destructively examined and confirm the tube degradation phenomena occurring at the tube support plates. No leakage during normal operating conditions has been observed at the support plates in the field at similar plants for crack indications with signal amplitudes less than 7.7 volts. Additionally, no primary-to-secondary leakage at the tube support plate has been detected in U.S. plants. Relative to the expected leakage during accident condition loadings, the limiting event with respect to primary-to-secondary leakage is a postulated steam line break event. Laboratory data for pulled tubes from other plants and model boiler specimens show limited leakage for indications under 10.0 volts during a postulated steam line break (SLB) condition (see Section 9.0 of WCAP-13187).

Additional Considerations

The proposed amendment would preclude approximately 10 manrem occupational radiation exposure that would otherwise be incurred by plant workers involved in tube plugging or repair operations. The proposed amendment would minimize the loss of margin in 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 those required for operation at full power. Reduction in the amount of tube plugging 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, thereby minimizing airborne contamination and exposure.

In addition, we will perform 100% bobbin coil inspection of the hot leg tubes to identify new areas of concern that may arise by providing a level of inservice inspection that is far in excess of the current T/Ss requirements.

NO SIGNIFICANT HAZARDS ANALYSIS

We have evaluated the proposed T/S changes and have determined that they do not represent a significant hazards consideration based on the criteria established in 10 CFR 50.92(c). Operation of the Cook Nuclear Plant in accordance with the proposed amendment will not:

1) <u>Involve a significant increase in the probability or consequences of an accident previously evaluated</u>

Testing of model boiler specimens for free standing tubes at room temperature conditions show burst pressures in excess of 5,000 psi for indications of ODSCC with voltage measurements as high as 19 volts. Burst testing performed on pulled tubes from other plants with up to 10 volt indications show burst pressures in excess of 5,900 psi at room temperature. 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 RG 1.121 criteria, requiring the maintenance of a margin of three times normal operating pressure differential on tube burst if through-wall cracks are present. Based on the existing data base, this criteria is satisfied with bobbin coil indications with signal amplitudes less than 6.8 volts, regardless of the indicated depth measurement. This structural limit is based on a -95% lower tolerance limit (LTL) confidence level of the data. The 1.0 volt plugging criteria compare favorably with the structural limit considering expected growth rates of ODSCC at Cook Nuclear Plant Unit 1. Alternate crack morphologies can correspond to 6.8 volts so that a unique crack length is not defined by a burst-pressure-to-voltage correlation. However, relative to expected leakage during normal operating conditions, no field leakage has been reported at other plants from tubes with indications with a voltage level of under 7.7 volts.

Also, a qualitative assessment is made between the beginning-of-cycle (BOC) 1.0 volt tube plugging criteria and the current 40% allowable tube wall penetration plugging criteria at Cook Nuclear Plant Unit 1. An ODSCC-degraded tube support plate intersection with a 4.0 volt bobbin coil response is expected to burst at approximately 7,400 psi, using the mean curve of Figure 9-2 of WCAP-13187. While the -95% LTL curve is used in the application of the plugging criteria, the mean curve must be used for this specific comparison in order to adequately compare the two data sets used. Per WCAP-13187, a comparison of the material properties at 650°F and room temperature condition properties showed that the elevated temperature properties are approximately 0.86 of the room temperature properties. Therefore, the temperature-adjusted burst pressure for a 4.0 volts bobbin coil indication is expected to be approximately 6,400 psi.

Figure 11 of NUREG-0718 plots the burst pressures of thinned 0.875 x 0.050 inch steam generator tubes. At 40% actual uniform wall thinning, extending 0.75 inch in axial length, the burst pressure is 6,800 psi. The NUREG test data is obtained at a temperature of 600°F, compared to the Westinghouse data noted above, which is adjusted for 650°F. The NUREG results at 40% actual thinning are comparable to the 4.0 volts BOC criteria expected burst pressure (6,800 psi versus 6,300 psi). The burst pressure for non-thinned tubes with partial depth cracks up to 0.75 inch in length is slightly lower than for uniform thinning up to depths of about 60%. Also, NUREG-0718 information can be used to estimate the burst pressure for a tube which has been slotted, simulating an axial crack. The expected burst pressure for a 40% deep, 0.75-inch long EDM slot using NUREG-0718 is approximately 6,000 psi. Therefore, it is judged that the margin of safety corresponding to the current 40% by NDE depth based plugging criteria is not significantly reduced upon implementation of a 4.0 volts bobbin coil criterion, which is higher than this proposed interim plugging criterion of 1.0 volt.

Relative to the expected leakage during accident condition loadings, the accidents that are affected by primary-to-secondary leakage and steam release to the environment are: feedwater system malfunction, loss of external electrical load and/or turbine trip, loss of all AC power to station auxiliaries, major secondary system pipe failure, steam generator tube rupture, reactor coolant pump locked rotor, and rupture of a control rod drive mechanism housing. Of these, the major secondary system pipe failure is the most limiting for Cook Nuclear Plant Unit 1 in considering the potential for off-site doses. Upon implementation of the interim plugging criteria, it will be verified that the distribution of cracking indications at the tube support plate intersections is such that primary-to-secondary leakage would result in site boundary doses within a small fraction of the 10 CFR 100 guideline, i.e., 30 rem thyroid, during a postulated SLB event. Data indicates that a threshold voltage of 2.8 volts would result in through-wall cracks with the potential to leak at SLB conditions. Application of the proposed plugging criteria requires that the current distribution of number of indications versus voltage be obtained. The indicated bobbin coil voltage is then combined with the rate of change in voltage measurement to establish an end-of-cycle (EOC) voltage distribution and, thus, leak rate during SLB pressure differential. If it is found that the projected SLB leakage for degraded intersections planned to be left in service exceeds 120 gpm, then additional tubes will be plugged to reduce projected SLB leakage below 120 gpm. Monte Carlo analyses results based on the Cook Nuclear Plant Unit 1 growth rate and assumed eddy current uncertainties indicate that over 4,000 indications, all with a (BOC) bobbin coil voltage of 2.0 volts, would contribute less than 1 gpm leakage at SLB conditions. Based on the inspection results from the last outage (1992), indications left in service are expected to have a total predicted SLB leak rate of 0.0 gpm at EOC conditions.

2) <u>Create the possibility of a new or different kind of accident from any previously analyzed</u>

Implementation of the proposed amendment does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism that could result in an accident outside of the region of the tube support plate elevations. Neither a single nor multiple tube rupture event would be expected in a steam generator in which the plugging criteria has been applied (during all plant conditions). The bobbin coil signal amplitude plugging criteria is established such that neither operational leakage nor excessive leakage during a postulated SLB condition are anticipated.

Indiana and Michigan Power Company will 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 upon application of the interim plugging criteria. The current technical specification limit on primary-to-secondary leakage at operating conditions is a maximum of 1.0 gpm (1440 gpd) for all steam generators or a maximum of 500 gpd for any one steam generator. 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. The 150 gpd limit provides 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 (Item 3 of Section 3.2 of WCAP-13187) 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 three against bursting at normal operating pressure differential. A voltage amplitude of 6.8 volts for typical ODSCC corresponds to meeting this tube burst requirement at the -95% LTL uncertainty limit on the burst correlation. Alternate crack morphologies can correspond to 6.8 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.

The single through-wall crack lengths that result in tube burst at three times normal operating pressure differential and SLB conditions are about 0.44 inch and 0.84 inch, respectively. Nominal leakage for these crack lengths would range from 0.1 gpm to 4 gpm, respectively, while lower 95% confidence level leak rates would range from about 0.01 gpm to 0.5 gpm, respectively.

An operating leak rate limit of 150 gpd will be implemented in application of the interim tube plugging criteria. This leakage limit provides for detection of 0.4 inch long cracks at nominal leak rates and 0.6 inch long cracks at the -95% LTL confidence level leak rates. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for SLB conditions at leak rates less than a -95% LTL confidence level and for three times normal operating pressure differential at less than nominal leak rates.

3) Involve a significant reduction in a margin of safety

The use of the interim plugging criteria for the tube support plate at Cook Nuclear Plant Unit 1 is demonstrated to maintain steam generator tube integrity commensurate with the requirements 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 of 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. Tubes with unacceptable cracking will be removed from service. The most limiting effect would be a possible increase in leakage during a steam line break event. Once the interim plugging criteria is applied, excessive leakage during a steam line break event is precluded by verifying that the expected end-of-cycle distribution of crack indications at the tube support plate elevations would result in minimal and acceptable primary-to-secondary leakage during all plant conditions. This helps to demonstrate that radiological conditions are less than a small fraction of the 10 CFR 100 guideline.

In addressing the combined effects of a loss-of-coolant accident (LOCA) and a 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. 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. 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 system primary loops, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design basis of the plant. Excluding breaks in the RCS primary loops, the LOCA loads from the large branch line breaks were analyzed for Cook Nuclear Plant Unit 1 and were found to be of insufficient magnitude to result in steam generator tube collapse or significant deformation.

Regardless of whether or not leak-before-break is applied to the primary loop piping at Cook Nuclear Plant Unit 1, any flow area reduction is expected to be minimal (much less than 1%) and PCT margin is available to account for this potential effect. Analyses results show that no tubes near wedge locations are expected to collapse or deform to the degree that secondary-to-primary in-leakage would be increased over current expected levels. For all other steam generator tubes, the possibility of secondary-to-primary leakage in the event of a combined LOCA and SSE event is not significant. In actuality, the amount of secondary-to-primary leakage in the event of a combined LOCA and SSE is expected to be less than that currently allowed, i.e., 500 gpd per steam generator. Furthermore, secondary-to-primary in-leakage would be less than primary-to-secondary leakage for the same pressure differential since the cracks would tend to close under a secondary-to-primary pressure differential. Also, the presence of the tube support plate is expected to reduce the amount of in-leakage.

Addressing RG 1.83 considerations, implementation of the interim plugging criteria is supplemented by 100% inspection requirements at the tube support plate elevations having ODSCC indications, reduced operating leak rate limits, and eddy current inspection guidelines to provide consistency in voltage normalization.

As noted previously, implementation of the interim plugging criteria will decrease the number of tubes which must be repaired or taken out of service by plugging. The installation of steam generator tube plugs reduces the RCS flow margin and, 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 change 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 Technical Specifications.

CONCLUSION

Based on the preceding analysis, it is concluded that using the tube support plate elevation bobbin coil signal amplitude interim steam generator tube plugging criteria for removing tubes from service at Cook Nuclear Plant Unit 1 is acceptable and the proposed license amendment does not involve a Significant Hazards Consideration as defined in 10 CFR 50.92.