PRIORITY 1
ACCELERATED RIDS PROCESSING

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9502130196 DOC.DATE: 95/02/03 NOTARIZED: NO DOCKET #
FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana M 05000315
AUTH.NAME AUTHOR AFFILIATION
FITZPATRICK,E. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

SUBJECT: Application for amend to license DPR-58 re TS changes to incorporate 2.0 volt interim steam generator tube support plate plugging criterion for fuel cycle 15.

DISTRIBUTION CODE: A001D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 13+12
TITLE: OR Submittal: General Distribution

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD3-1 LA	1	1	PD3-1 PD	1	1
	HICKMAN,J	1	1			
INTERNAL:	FILE CENTER 01	1	1	NRR/DRCH/HICB	1	1
	NRR/DSSA/SPLB	1	1	NRR/DSSA/SRXB	1	1
	NUDOCS-ABSTRACT	1	1	OGC/HDS2	1	0
EXTERNAL:	NOAC	1	1	NRC PDR	1	1

P R I O R I T Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 504-2083) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 11 ENCL 10

Indiana Michigan
Power Company
P.O. Box 16631
Columbus, OH 43216

INDIANA
MICHIGAN
POWER

February 3, 1995

AEP:NRC:1166Q

Docket No.: 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 constitute an application for amendment to the technical specifications (T/Ss) of Donald C. Cook Nuclear Plant Unit 1. Specifically, we request that steam generator interim 2 volt plugging criteria, approved for fuel cycle 14 by license amendment No. 178, dated March 15, 1994, be continued for fuel cycle 15. The cycle 15 outage is scheduled to begin in September 1995; therefore, we request approval of this amendment request prior to that time to avoid outage delays.

This T/S change is essential to support Cook Nuclear Plant Unit 1 continued operation while the NRC continues to review industry comments prior to formalizing draft Generic Letter 94-XX, "Voltage Based Repair Criteria for the Repair of Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," Federal Register 41520, August 12, 1994, and Advanced Notice of Proposed Rulemaking "Steam Generator Tube Integrity for Operating Nuclear Power Plants," Federal Register 47817, September 19, 1994.

Attachment 1 provides a technical summary of the requested changes as well as 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. No technical changes were made to the interim plugging criteria T/Ss used for fuel cycle 14. Changes were made only to reference fuel cycle 15 in the appropriate T/S paragraphs where fuel cycle 14 was previously referenced. Attachment 2 contains existing T/S pages marked to reflect the requested changes. Attachment 3 provides the proposed revised pages.

A001
1/1

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 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.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 3rd DAY OF February 1995

Rita D Hill
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
J. R. Padget

ATTACHMENT 1 TO AEP:NRC:1166Q

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 Cook Nuclear Plant Unit 1 steam generators T/Ss 4.4.5, 4.4.5.5, and 3.4-6.2, to allow continued use of steam generator tube support plate interim plugging criteria (IPC) for fuel cycle 15. The change allows steam generator 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 3.6 volts may remain in service if a rotating pancake coil probe reactor coolant system (RPC) inspection does not detect degradation.

This amendment, specific to fuel cycle 15, would reduce the number of steam generator tubes plugged due to indications at support plate intersections. Reducing the number of plugged tubes provides ALARA benefits and increases RCS flow margin.

The proposed T/S change maintains for fuel cycle 15 the same technical basis for IPC utilized for fuel cycle 14, previously discussed in submittal documents AEP:NRC:1166H, 1166L, and 1166M and the NRC safety evaluation associated with license amendment No. 178, dated March 15, 1994. Eddy current inspection, reporting, and leakage requirements pertinent to fuel cycle 14 will be maintained for fuel cycle 15.

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. The maximum projected EOC 14 voltage 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.

The proposed changes are those necessary to incorporate references to fuel cycle 15 in T/S 4.4.5.2.e, 4.4.5.5.e, 3.4.6.2.c and Bases 3/4.4.5 and 3/4.4.6.2.

10 CFR 50.92 EVALUATION

BACKGROUND

Cook Nuclear Plant Unit 1 T/S Amendment 178 permitted the implementation of a 2.0 volt steam generator tube IPC for the 14th operating cycle of the Cook Nuclear Plant Unit 1 steam generators. This license amendment was applicable only for the previous cycle (cycle 14), and required the repair of flaw-like bobbin indications above 2.0 volts. We are proposing use of a 2.0 volt interim repair criterion for the upcoming cycle 15.

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 IPC 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 14 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 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 IPC. Per draft NUREG-1477, all bobbin indications are included in the steamline break leakage analyses along with the consideration of probability of detection. 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 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 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 EOC 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 potential throughwall cracks. 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 14 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 loss of coolant accident (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 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 notches 0.75 inch long, provided the tube support plate is adjacent to the notched area. Because tube to tube support plate proximity precludes tube burst during normal operating conditions, use of the criteria must retain tube integrity characteristics which maintain the RG 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 beginning of cycle [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 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 x RL) + (0.4 x RL) = 9.6 volts, or,

the maximum allowable BOC repair limit can be expressed as,

RL = 9.6 volt structural limit/1.6 = 6.0 volts.

This structural repair limit supports this application for cycle 15 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 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 steamline break

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 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 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. Projected EOC 14 leak rates were calculated 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 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 2.0 volt IPC during cycle 15 would 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 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; 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 15 T/S limits imposed on primary to secondary leakage at operating conditions are: 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 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.

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 general design criteria 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 15 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 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. 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 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 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.