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SUBJECT: Forwards addl info requested re 950804 proposed TS changes re hybrid expansion joint sleeved tube repair boundary limits,in response to 950814 followup telcon & 950901 RAI.

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AEP:NRC:1129G

Docket No.: 50-315

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

October 23, 1995

Gentlemen:

Donald C. Cook Nuclear Plant Unit 1
REQUEST FOR ADDITIONAL INFORMATION:
TECHNICAL SPECIFICATION CHANGES REGARDING
HYBRID EXPANSION JOINT SLEEVED TUBE REPAIR BOUNDARY LIMITS

Our letter AEP:NRC:1129E, dated August 4, 1995, proposed technical specification (T/S) changes regarding hybrid expansion joint sleeved tube repair boundary limits. Subsequent to the submittal, a meeting was held with the NRC staff at the NRC offices in White Flint, MD, a followup teleconference was held on August 14, 1995, and a request for additional information (RAI) dated September 1, 1995, was received.

The attachments to this letter provide additional information requested to be placed on the docket by the NRC staff. Attachment 1 contains a probabilistic risk assessment evaluation of the proposed hybrid expansion joint sleeved tube repair boundary limits, and Attachment 2 contains responses to three additional questions regarding the analyses supporting the proposed amendment. Attachment 3 contains the response to the September 1, 1995, RAI.

Our original hybrid expansion joint (HEJ) submittal stated that 0.033 gpm will be allocated for each HEJ indication towards the steam line break accident steam generator tube leakage limitation of 12.6 gpm (currently contained in the Bases for T/S 3/4.4.6.2). The 12.6 gpm limit was originally added as part of the 2.0 volt interim plugging criteria, which was most recently renewed in Amendment 200. The 12.6 gpm was based on limiting offsite doses in the event of a steam line break to within 10 percent of 10 CFR 100 guidelines. We have recently been informed by Westinghouse of an error in their offsite dose calculations, which resulted in the need to reduce the calculated limit from 12.6 gpm to 8.4 gpm. This error has no impact on the current cycle, because the projected steam line break leakage is significantly below the limit.

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The letter from Westinghouse notifying us of the error is included as Attachment 4. Because the revised limit is in the conservative direction, the error does not impact our previous no significant hazards determination. A revised Bases page which incorporates the revised limit, as well as other pages reconciled with overlapping amendments, will be provided to the NRC when the HEJ amendment is ready for issuance.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 23rd DAY OF October 1995

Rita W Hill
Notary Public

My Commission Expires: 6-28-99

eh

Attachments

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

ATTACHMENT 1 TO AEP:NRC:1129G

PRA EVALUATION OF PROPOSED HYBRID EXPANSION JOINT SLEEVED TUBE REPAIR BOUNDARY LIMITS

PRA EVALUATION OF PROPOSED HYBRID EXPANSION JOINT SLEEVED TUBE REPAIR BOUNDARY LIMITS

Background

The Donald C. Cook Nuclear Plant unit 1 currently has 1,839 steam generator tubes with sleeves. Recently, industry experience indicates that parent tubes into which these sleeves are inserted are susceptible to degradation in the region of the sleeve hybrid expansion joint (HEJ). Based on deterministic analysis, this requested technical specification amendment effectively requests that repair boundary now be expanded to the HEJ lower hard roll transition. The HEJ relies on the contact pressure in the hard roll region, so circumferential degradation below the hard roll has no impact on the tube integrity.

Even if one were to postulate failure, Westinghouse has provided justification that, should the HEJ fail, the tube is restrained by the nearby tubes such that the tube cannot lift clear of the sleeve. This results in far less flow than a tube rupture.

Over 28,000 sleeves of this or similar design have been installed throughout the industry, although only 12,000 tubes are currently in service. No sleeves have burst, and only one has had a significant leak due to a crack at the HEJ upper hard roll transition. The Cook Nuclear Plant unit 1 sleeves were installed in 1992.

PRA Evaluation

Questions:

1) What is the increase in steam generator tube rupture (SGR) initiating event frequency?

2) What is the increased potential for induced tube ruptures?

3) What is the increased potential for multiple tube ruptures?

4) Given this information, what is the impact on core damage frequency and offsite releases.

Responses:

1) Increased potential for steam generator tube rupture

The current PRA initiating event frequency is based on Westinghouse data of seven tube failures over 1.55E7 tube-years of operation. The sleeve alloy, Alloy 690, is superior to the original tubing,

particularly with respect to corrosion. Degradation of the parent tube in the HEJ hard roll area is not allowed for the tube to remain in service, and regular tube inspections will confirm joint adequacy. Due to physical restraint of most of the tubes by adjacent tubes, the consequences of failure of the joint in a manner causing the tube to slip upwards would be limited to a leak and not the full flow of a tube rupture. Therefore, an increase in steam generator tube rupture frequency or consequences is not expected.

To conservatively bound a potential increase in tube rupture frequency, the requested changes in the joint boundary acceptance criteria can be treated as a new tube design. Previous generic failure experience on previous tube designs is represented by steam generator tube failure data used for the overall tube failure rate. If the HEJ is conservatively treated as an additional source of tube failure, and the industry tube failure data is conservatively applied to that single joint, the steam generator tube rupture initiating event frequency can be scaled by the ratio of the number of sleeves plus tubes to the number of tubes used in the base estimate. This results in an increase of 13% in steam generator tube rupture initiation frequency.

2) Increased potential for induced tube ruptures

A steam line break accident produces the greatest stress on the steam generator tubes. The HEJs have sufficient design margin to remain intact even in this situation. To induce ruptures, a degradation mechanism must be postulated. Induced failures would then occur in weakened tubes which have not yet progressed to failure during normal operation.

The Cook Nuclear Plant PRA does not currently consider steam generator tube failures induced by steam line breaks. Previous studies have found about a three percent to nine percent induced failure rate for degraded tubes. Even though no degradation mechanism has been postulated associated with the technical specification change, it can be reasonably assumed that this hypothetical degradation mechanism could be represented by an average six percent failure rate for four steam generators. Using the initiating event frequency of a steam line break and the fraction of sleeved tubes, the initiation frequency of a steam line break-induced steam generator tube failure is estimated to be about 1.0E-6/yr. Since the emergency operating procedures address this event, estimates of operator action failure rates and equipment reliability would provide a core damage frequency estimate of less than a negligible 1.0E-7/yr.

3) Increased potential for multiple tube ruptures

For an induced single tube rupture, at least one tube must be susceptible to the pressure differential produced by a steam line break. For multiple ruptures, more than one tube must be susceptible. Per the previously cited reference, this is a factor of ten less probable than the single induced failure. Therefore, even the initiation frequency due to multiple failures is less than 1.0E-7/yr.

4) *Impact on core damage frequency and offsite releases.*

Since the proposed technical specification change will not change the structural design criteria of the steam generator tubes, no impact is expected on the probability of a steam generator tube rupture (either at normal operating conditions or induced by a severe transient), nor on offsite releases. Conservative estimates of a potential increase in the steam generator tube rupture susceptibility indicated an increased core damage frequency of less than 1.0E-6/yr, and a best estimate increase would be expected to be significantly less than this value.

ATTACHMENT 2 TO AEP:NRC:1129G

RESPONSE TO ADDITIONAL QUESTIONS
REGARDING HYBRID EXPANSION JOINT
SLEEVED TUBE REPAIR BOUNDARY LIMITS ANALYSIS

This attachment contains responses to three additional questions raised by the NRC regarding the analyses supporting the hybrid expansion joint (HEJ) sleeved tube repair boundary limits.

QUESTION 1: Provide details regarding the establishment of the 0.06 uncertainty applied to the location characterization of indications with reference to the bottom of the hard roll upper transition.

RESPONSE: In our letter AEP:NRC:1129E, information was provided which described the theoretical uncertainty involved for a surface-riding pancake probe with coil diameters of 0.080 inch. The uncertainty is defined by the square root of the sum of the squares of the coil diameters, or 0.06 inch. This uncertainty can be applied to the location characterization of parent tube flaws. The practical application of the probe and analyst techniques can also affect apparent uncertainty.

Test data supporting the 0.06 inch nominal uncertainty is provided in a proprietary Westinghouse report, 87-5D4-CYUDS-R1. Summarizing, this report provides data which shows that for 24 test samples with indications ranging from 0.23 to 3.10 inches from the bottom of the roll expansion transition of a steam generator tube, the measured uncertainty varied from 0.05 to 0.07 inch. This error includes the accumulated error of locating both the transition and indication using RPC. The associated geometry of a sleeved tube is judged not to affect these uncertainty values based on the analyst techniques described later. It should be noted that the uncertainty values of 0.05 to 0.07 inch were obtained using 1987 eddy current testing probes, analyst techniques and software. The advances in probe design, analyst sensitivity and practical experience, and evaluation software can only decrease the actual uncertainties experienced in the field.

Analyst Techniques:

Figure 1 shows RPC pancake coil data of a typical HEJ configuration in axial profile. The strip chart and the profile C-scan are labeled with the main features of the joint: the upper and lower transitions of the hydraulic expansion (HEUT and HELT, respectively), and the upper and lower transitions of the hard roll (HRUT and HRLT, respectively). All features of the HEJ are evident to the analyst. Figure 2 shows the same sleeved tube using a Plus Point probe.

The analyst will calibrate the speed of the probe using known features of a standard or, alternatively, the length of the hydraulic expansion. All measurements made by the analyst will be made relative to HRUT. The location of HRUT is denoted on Figure 3 as the low point on the profile of the HEJ. Figure 3 also shows an example of a measurement of the position of an indication made relative to HRUT. Any measurement of the location of an indication or the HRLT will be made relative to this one point. Thus, these measurements would maintain the same uncertainty cited previously. Figure 4 shows the same indication detected using a Plus Point coil. Figures 5 and 6 show the location of an indication in another HEJ.

The three Kewaunee pulled tubes were also RPC inspected in the laboratory using both the 3-coil RPC and Plus Point probes. Tubes R2 C32 and R2 C54 were destructively examined, and therefore actual crack location information can be obtained. For tube R2 C32, the 3-coil RPC indicated that the crack location was 1.02 inch below the bottom of the hard roll upper transition, while the Plus Point probe indicated the crack location was 0.98 inch below the bottom of the hard roll upper transition. The listed crack location from destructive examination is 1.00 inch. The 3-coil and Plus Point crack location measurements for R2 C54 are 1.11 inch and 1.23 inch, respectively. The crack location as determined by destructive examination is 1.2 inch. The conclusions which can be drawn from this limited sample are that the 3-coil and Plus Point would be equally effective at location characterization. While the Plus Point exhibited location error within 0.03 inch, the maximum error associated with the 3-coil was applied in the conservative direction, i.e., the 3-coil determined that the crack was higher than it actually was.

Impact of Roll Tolerances on Hard roll Flat Length

The flat length of the roller is a nominal dimension of one inch. Since the rollers are heat treated to an extremely hard condition, manufacture into the final shape is done by grinding. The tolerances involved are held to a few thousands of an inch; therefore, the flat length of the roll expanded zone should only vary by a few thousands of an inch from tube to tube.

QUESTION 2: Discuss crack growth rate data obtainable from the past two Kewaunee sleeved tube inspections.

RESPONSE: During the recent Kewaunee HEJ sleeved tube inspections, a sample of HEJ sleeved tubes was inspected using the I-coil probe, which was used for the 1994 inspection. Eddy current data for the 1995 I-coil inspections were compared to the 1994 data and results. A total of only 33 tubes were compared, due mainly to the large number of tubes with identified indications in both 1994 and 1995. Of these 33 comparisons, two were NDD, two had noisy data and could not be evaluated, and the other 29 all indicated circumferential indications in both 1994 and 1995. This suggests that the cracking occurring at Kewaunee is not rapidly progressing and has been occurring for several cycles. The 1994-1995 I-coil data was evaluated for arc length growth of indications. The average arc length growth was determined to be nine degrees.

Additionally, the 1994 I-coil data for the three pulled HEJ sleeved tubes were re-evaluated. Only two of the indications had retrievable data. Re-evaluation of the data indicates greater than 270 degree indications in the 1994 data. Based on the 1995 destructive examination data which shows average macrocrack depth of 60% throughwall and maximum depth of 92% throughwall, with minimum corrosion depths as low as 5%, and when considering the reevaluation of the 1994 inspection results indicating circumferential degradation greater than 270 degrees, it can be inferred that crack growth for these tubes during the last operating cycle was small. Similar conclusions can be drawn as for the 33 indications discussed above, i.e., the observed cracking is not progressing at a rapid rate.

QUESTION 3: Provide information related to the concentration capability of contaminant species in the HEJ upper hydraulic and hard roll upper transitions.

RESPONSE: Chemistry Concerns Related to Concentration of Contaminants in the Crevice Region

The results of the destructive examination of the Kewaunee pulled HEJ sleeved tubes indicated the presence of boron and lithium in the crack faces,

suggesting the presence of primary water. All tests indicated that the environment was not chemically aggressive and the pH of the crevice solution was determined to be approximately 7.1. Since it is believed that PWSCC is predominantly stress driven, the presence of the boron and lithium merely suggests that a primary water environment and substantial residual stresses were the main contributors to the observed cracking.

Recent EPRI sponsored investigations have shown that crack growth rates of Alloy 600 tubing are independent of coolant pH and expected normal RCS lithium concentrations. Statistical evaluation of existing corrosion test data indicates that tube residual stresses and material properties dominate crack initiation due to PWSCC while normal RCS chemistries represent a second order influence. Additional testing included variances in dissolved hydrogen levels, pH, and lithium concentrations, again supporting the conclusion that the causative mechanism in PWSCC is dominated by residual stress.

An assessment of the potential for development of an aggressive environment in the upper transitions has been evaluated and determined to be negligible. Boiling or local dryout cannot occur at the upper hydraulic transition or at the hard roll upper transition due to the pressure of the RCS, which is well above the saturation vapor pressure of the coolant. Additionally, the short (1/2 inch) crevice formed at the non-expanded end of the sleeve would provide for good communication with the primary coolant flow, preventing fluid stagnation.

Residual Stress Concerns in the HEJ Hard roll Transitions

Calculations performed by Westinghouse indicate that the residual ID axial stresses in typical hard roll transitions tend to be higher at the lower transition during operation by 1 to 5 ksi. This stress difference may not always be large enough to clearly identify installation and thermal stresses solely as the causative mechanism for cracking in the lower transition versus the upper transition. However, the pulled tubes show no evidence of cracking, either circumferential or axial, at the upper hydraulic or hard roll upper transitions, even after a 10,000 lb axial load was applied to the tube. The field NDE results indicate that for the 938 circumferential

indications detected to date, approximately 99% have been identified in either the hard roll lower transition or lower hydraulic transition. There were no observations of cracking both above and below the hard roll in the same tube. From these observations, it is evident that an additional contributing mechanism is biasing the initiation of PWSCC in the lower transitions. Additional stress mechanisms may be due to the mechanical rolling action. While unsubstantiated, the most likely mechanism is believed to be the rolling action. The action of mechanical roll expansion results in significant radial preloads after the initial forward direction roll. In order to remove the roller from the tube, it must be directionally reversed. Without a sufficient impact breakaway force applied to the tapered mandrel driving shaft, the residual radial preload keeps the roller in engagement with the shaft, and can result in "roll down," which is essentially a partial mechanical roll expansion with the roller turning in the reverse direction as extraction of the tooling occurs. This would suggest that a small amount of roll down is present in every roll expanded joint. The localized impact upon tube residual stresses is unclear at this point but reasonable to consider as an additional contributing mechanism. It is also believed that cracking at one location can have a "stress relieving" effect at other elevations in the tube, as implied by the field inspection results.

The detection threshold of the Cecco-5 probe used at Cook Nuclear Plant unit 1 during the recent Cycle 14-15 refueling outage has been established to be 50% depth for both circumferential and axial indications. (No plugging was required due to HEJ degradation.) Therefore, the inspection technique capability applied to the parent tube above the tube repair boundary is entirely consistent with the remaining sections of the tube and other non-sleeved tubes. The extended operating history and inspection history of HEJ sleeved tubes at Kewaunee and Point Beach 2 indicate that rapidly propagating indications at the upper transitions are not anticipated. Additionally, should significant indications occur at Cook Nuclear Plant during future cycles, the capability of the Cecco-5 probe to detect such indications has been proven. Throughwall degradation at the upper transitions would be expected to result in detectable levels of primary to secondary leakage. In summary, while evaluation of parent tube stress levels in the hard roll transitions cannot definitively identify a causative mechanism for

cracking in the lower transitions, the stress dependency of the cracking phenomena clearly shows a substantial difference in applied stress levels. The overwhelming statistical likelihood of degradation in the lower transitions, as opposed to the upper transitions also supports the conclusion that degradation in the parent tube at the hard roll lower transitions does not imply pending degradation at the upper transitions.

Figure 1

3-Coil Profile of HEJ with Parent Tube Indication

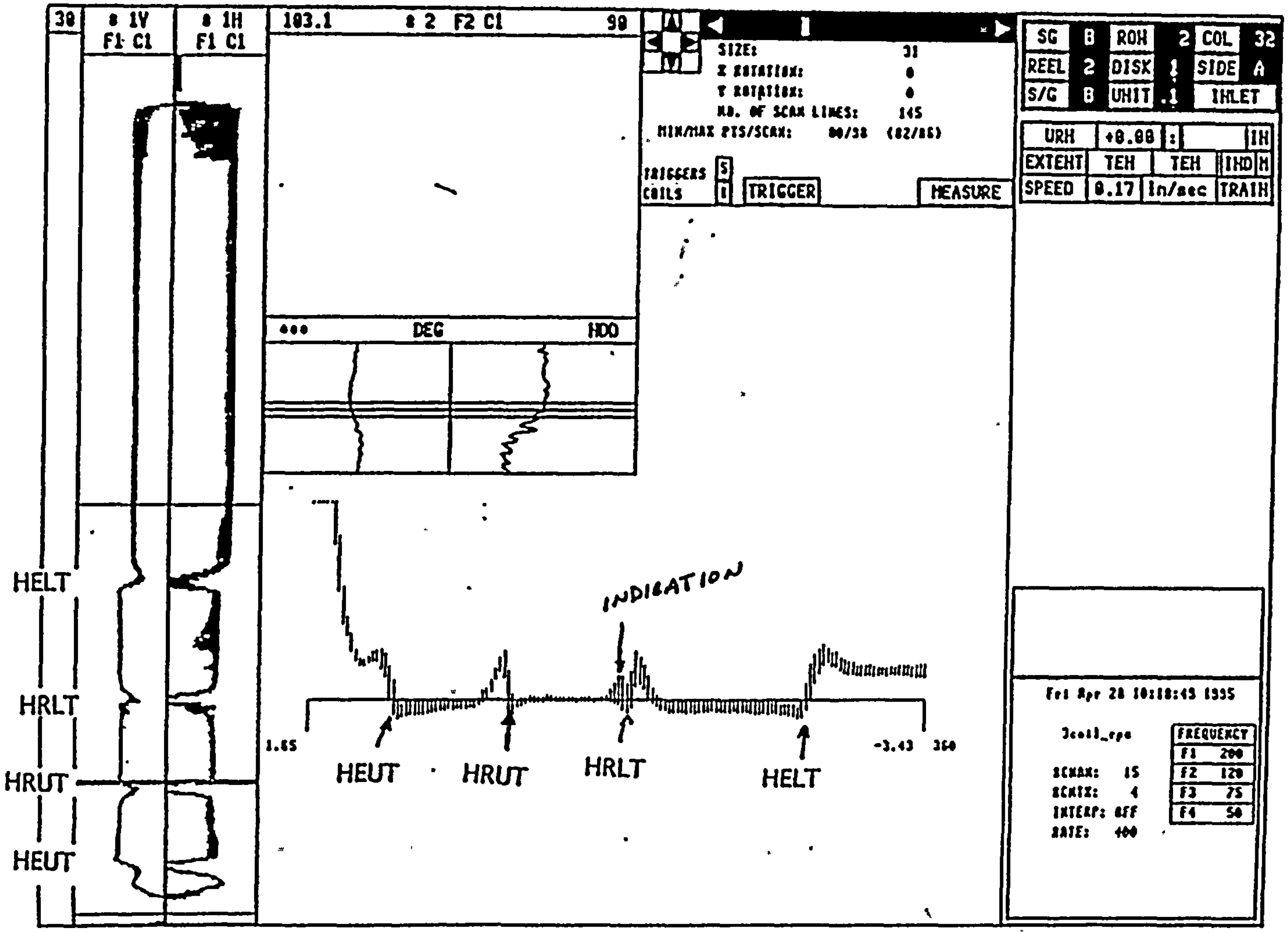


Figure 2

Plus Point Profile of HEJ with Parent Tube Indication

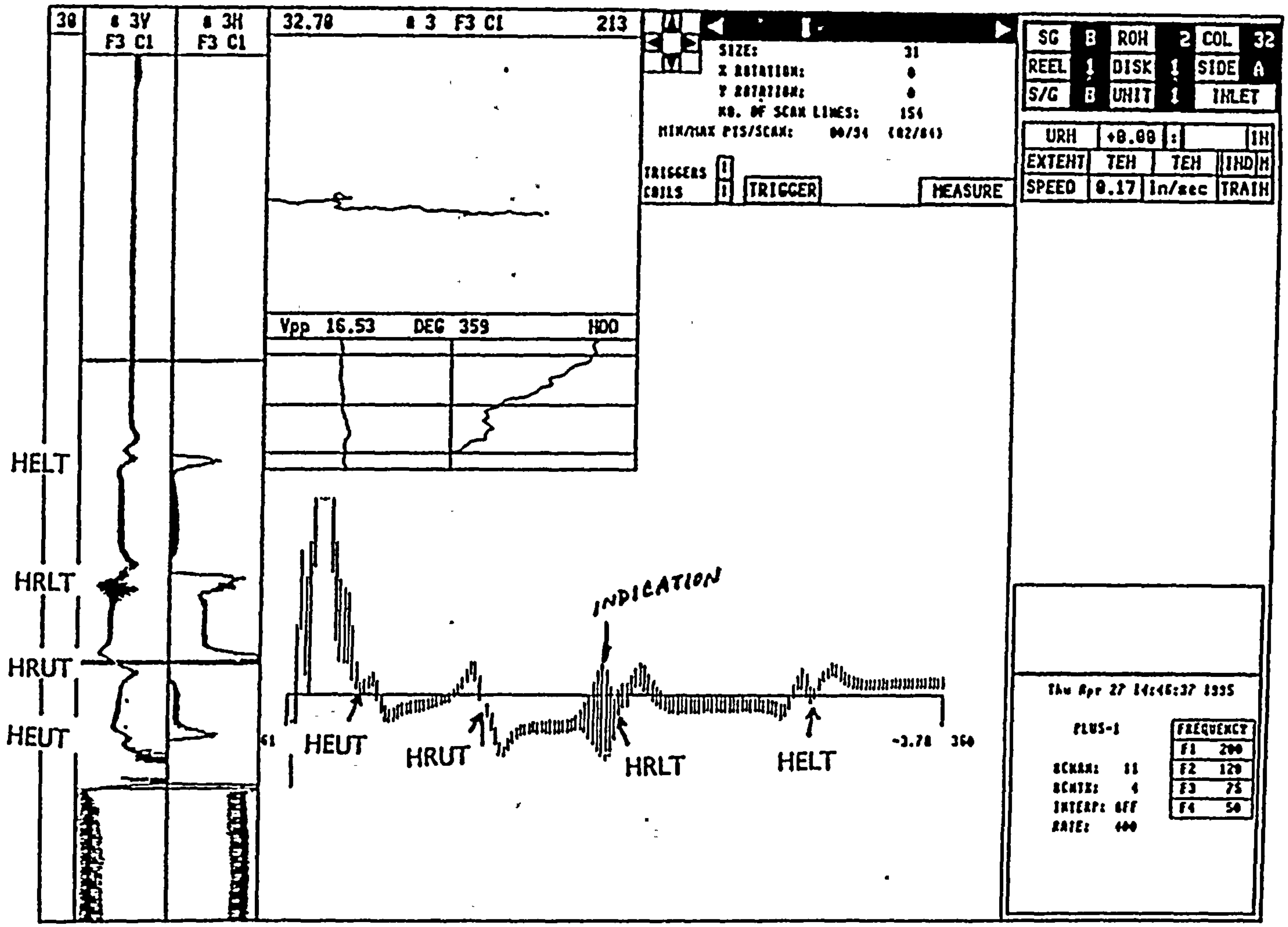


Figure 3

3-Coil Location of Indication Relative to HRUT: R2 C32

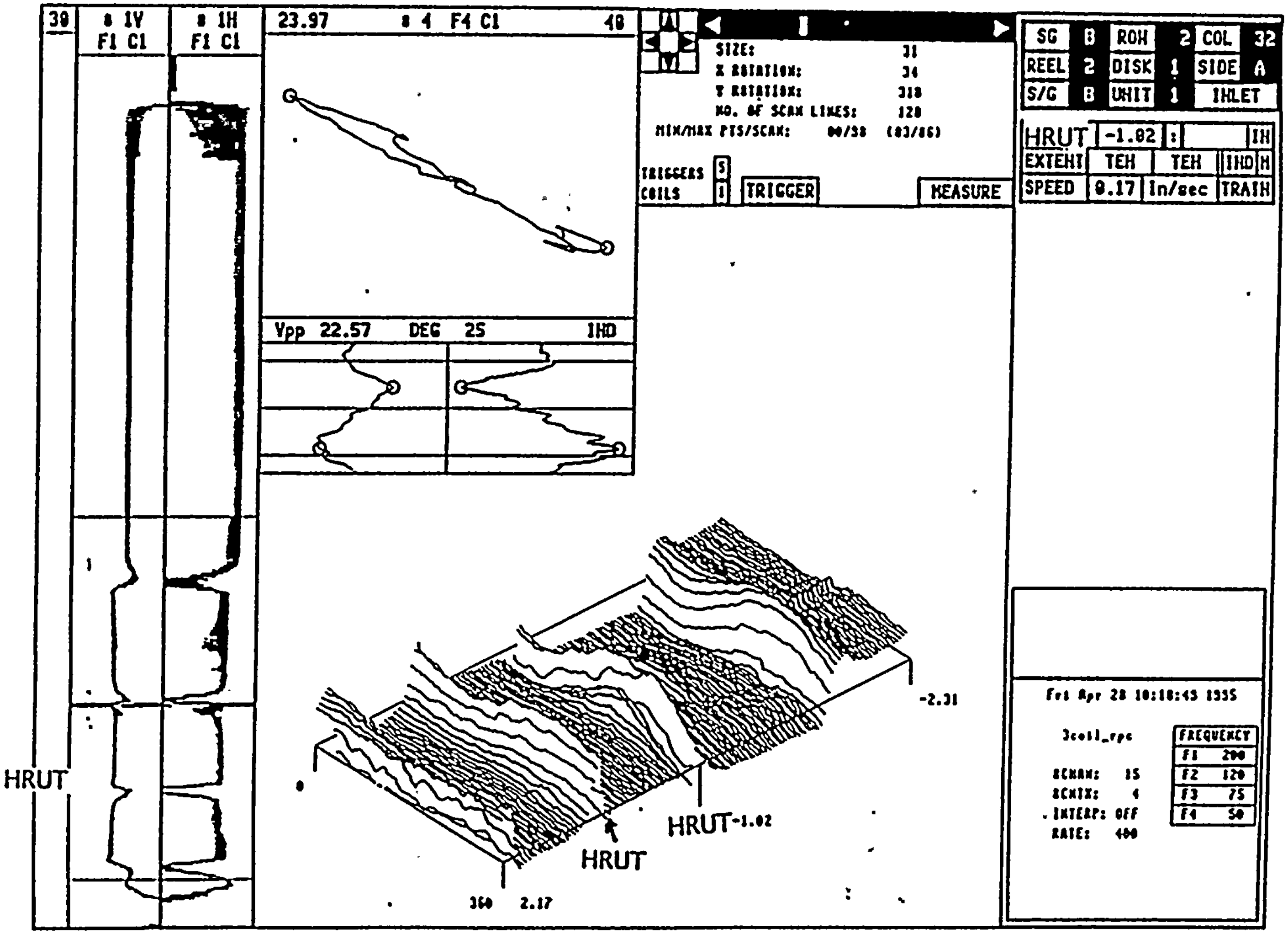


Figure 4

Plus Point Location of Indication Relative to HRUT: R2 C32

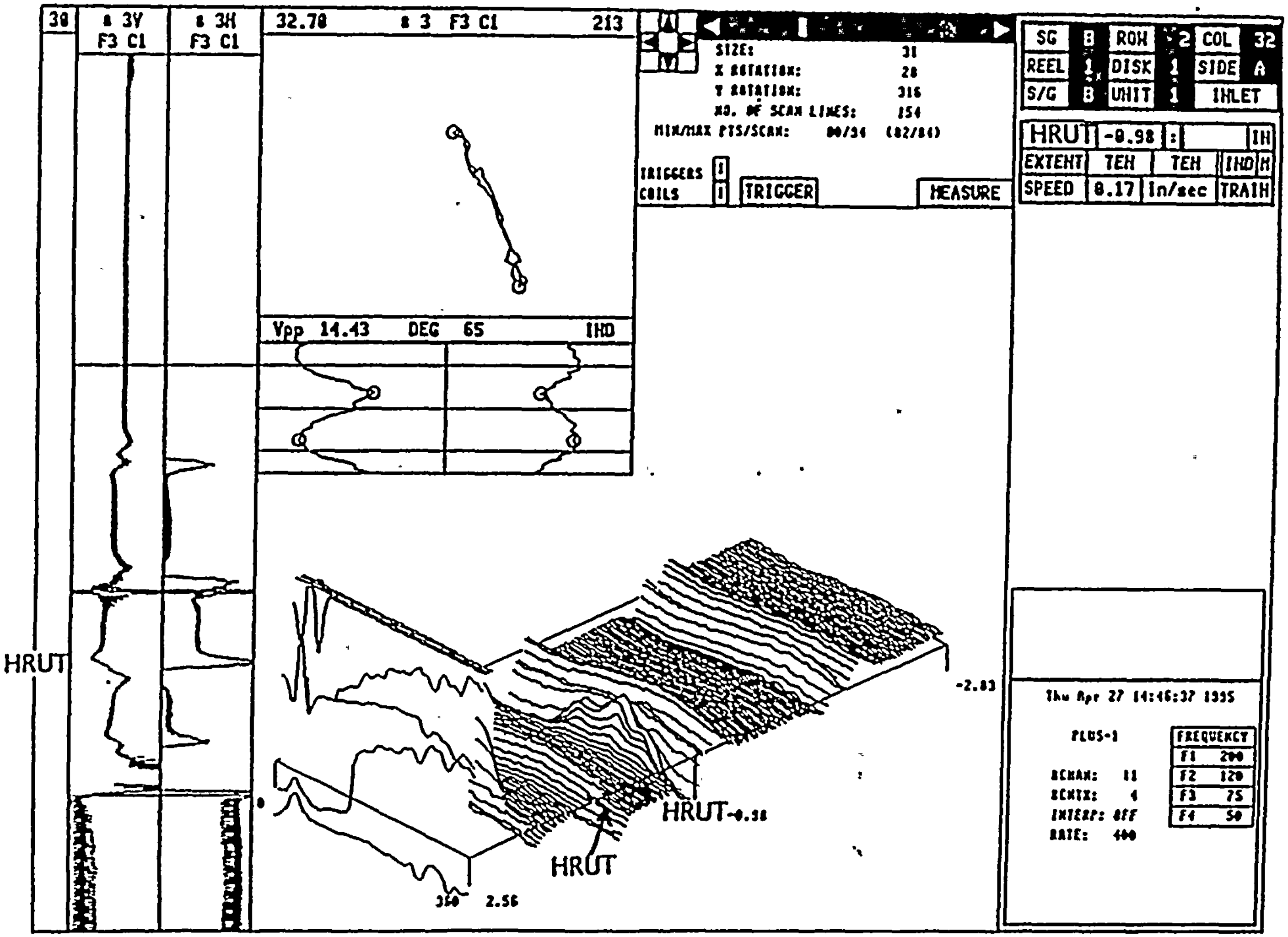


Figure 5

3-Coil Location of Indication Relative to HRUT: R2 C54

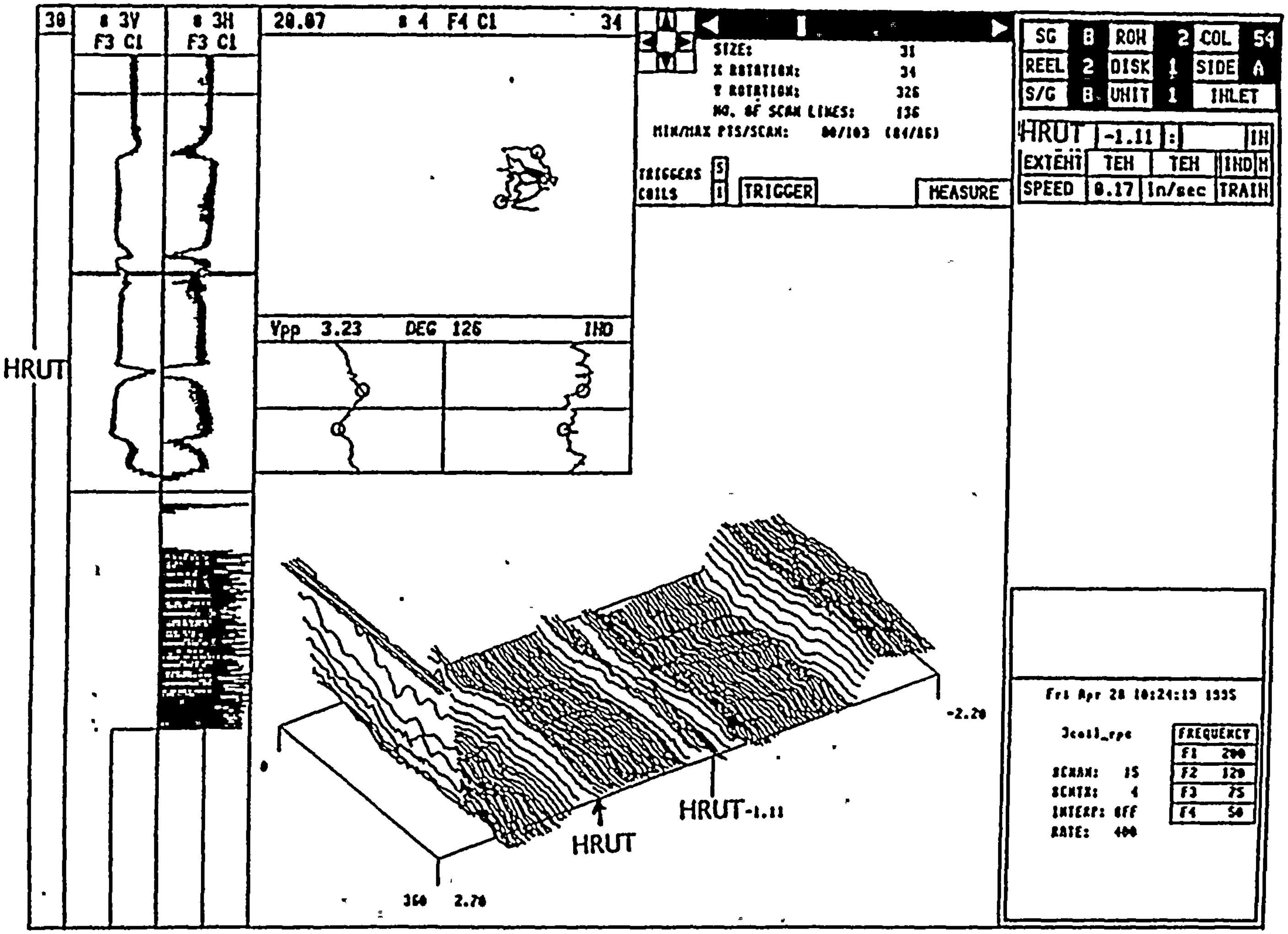
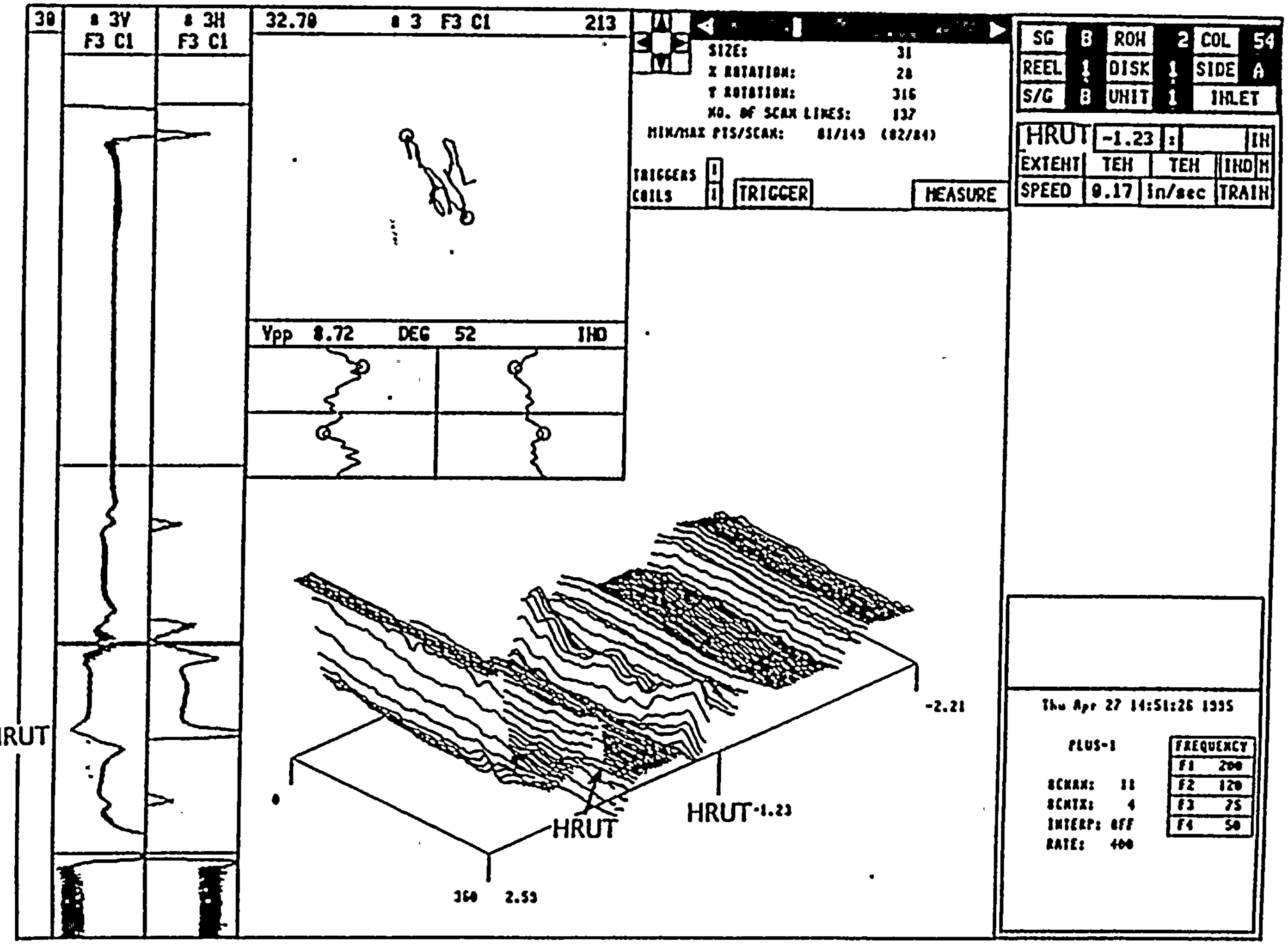


Figure 6

Plus Point Location of Indication Relative to HRUT: R2 C54



ATTACHMENT 3 TO AEP:NRC:1129G

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
DATED SEPTEMBER 1, 1995

A request for additional information dated September 1, 1995, requested the following:

QUESTION: Address the impact of plant operation with circumferentially cracked HEJ sleeved steam generator tubes, concentrating on the ability of the steam generators to perform their fission product containment function under high temperature and pressure conditions that could result from severe accidents.

RESPONSE: The impact on plant operation with a circumferentially cracked HEJ sleeved tube under any conditions, including severe accident scenarios, is mitigated by the HEJ sleeve design and the in-service inspection program for sleeved tubes.

Design: Installation of an HEJ sleeve restores a degraded tube's integrity by establishing a new pressure boundary. No structural credit is taken for the parent tube span between the lower and upper sleeve-to-tube joints. Therefore, circumferential indications in the parent tube in this region are no longer a concern since the sleeve now acts as the pressure boundary providing structural integrity and the barrier providing adequate leak tightness.

In-Service Inspections: Eddy current testing of HEJ sleeved tubes at the end of each operating cycle using enhanced techniques is readily capable of detecting circumferential cracking in the parent tube in the region of the HEJ sleeve. Detection of circumferential cracking outside the defined repair boundaries would require the sleeved tube to be plugged, thereby preventing circumferential indications in an active pressure boundary from being returned to service.

The effect of the proposed technical specification change on the probability of impacting core damage frequency and offsite releases was discussed previously, in Attachment 1. This evaluation concluded that, since the proposed technical specification change will not change the structural design criteria of the steam generator tubes, no impact is expected on either the probability of a steam generator tube rupture (either at normal operating conditions or induced by a severe transient), or on offsite releases.

During a severe accident with no primary system breach, as the core is melting but before the vessel fails (which relieves pressure) the steam generator tube could see both high differential pressure (as high as the pressurizer safety valve setpoints) and high gas temperatures (from the melting core). If there is no water on the secondary side, the combination of

pressure stress and high temperatures could allow the tubes to creep and rupture. This would lead to a containment bypass sequence, which would have greater offsite release consequences than the otherwise expected inside containment sequence.

The Cook Nuclear Plant PRA was reviewed for core damage sequences that result in the conditions where steam tube creep failure is a potential concern, sequences with high reactor coolant system pressure and no water on the secondary side of the steam generators. Sequences with these conditions occur with a frequency of about 2E-6/yr, contributing less than five percent to the total core damage frequency.

In the sequences that dominate the above set, the pressurizer power operated relief valves would still be available. Severe accident management guidelines are being implemented at Cook Nuclear Plant that will direct the operators to depressurize the reactor coolant system to prevent creep failure. Given an estimated 0.3 failure rate for this action, the potential for creep failure is reduced from 2E-6/yr to 6E-7/yr. The normal core damage frequency for the Cook Nuclear Plant PRA resulting in containment bypass is 9.7E-6/yr. Thus, if the HEJ is susceptible to creep failure, the increase in bypass frequency would be less than 10%, which is considered to be of marginal increased consequence.

ATTACHMENT 4 TO AEP:NRC:1129G

WESTINGHOUSE NOTIFICATION OF
REVISED ALLOWABLE PRIMARY-TO-SECONDARY
LEAK RATE DURING STEAM LINE BREAK

AEP-95-198

Westinghouse Electric Corporation

Energy Systems

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230-0355

Mr. Mark Ackerman
Nuclear Licensing and Fuels Section
American Electric Power Service Corporation
One Riverside Plaza
Columbus, OH 43215

NTD-NSRLA-OPL-95-477

October 18, 1995

AMERICAN ELECTRIC POWER SERVICE CORPORATION
DONALD C. COOK NUCLEAR PLANT UNIT 1
Revised Allowable Primary-to-Secondary Leak Rate During Steamline Break

Dear Mr. Ackerman:

As part of the Interim Plugging Criteria for the Cook Nuclear Plant Unit 1 steam generators, an analysis was performed to determine the allowable steam generator primary-to-secondary leak rate during a steamline break. The allowable leak rate was calculated to be 12.6 gpm in the faulted steam generator, assuming that the 0-2 hour site boundary (SB) thyroid dose would be limiting. Therefore, the 0-8 hour low population zone (LPZ) thyroid dose was not considered.

Subsequently, during the steamline break offsite dose analysis for the Cook Nuclear Plant Unit 2 uprating, it was determined by Westinghouse that the LPZ is limiting for Cook Nuclear Plant Unit 1. This necessitated a reanalysis of the allowable leakage. In addition to the salient assumptions used in the original analysis, the following assumptions were used to determine the LPZ thyroid dose:

- 978,000 lb. of steam released from the three intact steam generators from 2 to 8 hours
- LPZ X/Q of 7.5E-5 sec/m^3

The revised allowable leak rate was determined to be 8.4 gpm in the faulted steam generator. As in the original analysis, the leak rate in each of the three intact steam generators is 150 gpd (approximately 0.1 gpm).

AEP-95-198
NTD-NSRLA-OPL-95-477
October 18, 1995

The original value of 12.6 gpm and the revised value of 8.4 gpm for allowable leakage are both based on using the ICRP 2 (i.e., TID-14844) dose conversion factors (DCF). If the ICRP 30 DCFs are used, which are acceptable to the NRC, the allowable leakage is 15.1 gpm in the faulted steam generator.

In order to prevent a similar error, the Westinghouse analysts that perform these calculations have been notified of the issue. In addition, a Quality Performance Feedback (QPF) item was initiated. This QPF requires formal guidance addressing this issue to be provided by October 31, 1995. As part of this guidance, it will be required that both the SB and LPZ thyroid doses be calculated when determining the allowable primary-to-secondary leak rate following a steamline break for an IPC plant.

For Cycle 15, a Technical Specification change is not required, since there is no impact on any LCO and since the projected value for primary-to-secondary leakrate of 0.2 gpm is far below the revised allowable leakrate of 8.4 gpm in the faulted steam generator.

If you have any questions or need additional information, please call Ms. Robin Lapides (412/374-5683) or me.

Very truly yours,

Keith F. Matthews
Senior Sales Engineer
North American Field Sales

RSL/bbp

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