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#2
4/29

April 25, 1994

Mr. William Russell, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Braidwood Station Unit 1
Request for EMERGENCY TECHNICAL SPECIFICATION AMENDMENT
Facility Operating License: NPF-72
Technical Specification 3/4.4.5
NRC Docket No. 50-456

References: See Attachment G

Dear Mr. Russell:

Pursuant to 10 CFR 50.91(a)(5), Commonwealth Edison Company (CECo) proposes to amend Appendix A, Technical Specifications of Facility Operating License NPF-72, and requests that the Nuclear Regulatory Commission (NRC) grant an emergency amendment to Technical Specification 3/4.4.5 "Steam Generators."

The current Technical Specification 3/4.4.5 describes the necessary surveillance requirements and acceptance criteria to determine steam generator operability.

The proposed amendment modifies the Technical Specifications to incorporate a 1.0 volt steam generator tube interim plugging criteria for Cycle 5. The attached safety analysis shows that this proposal will have minimal impact on safety. Amending Technical Specification 3/4.4.5 would allow tubes with flaw indications at the tube support plates to remain in service regardless of depth provided they meet the following criteria:

A001 1/1 Change: NRC PDR 1 Ltr 1 Encl Want Prop

- Axial Outer Diameter Stress Corrosion Cracking (ODSCC) is the dominant degradation mechanism,
- Indications are within the tube support plate length,
- Indications are less than or equal to 1.0 volt as measured per the applicable inspection guidelines,
- Indications greater than 1.0 volt and less than or equal to 2.7 volts and are not confirmed by rotating pancake coil (RPC) probe inspection, and
- Indications are located outside of the "wedge" areas of the steam generators.

This philosophy is in general agreement with the Draft NUREG-1477, "Voltage-Based Interim Plugging for Steam Generator Tubes-Task Group Report" and is consistent with recent Safety Evaluations issued to Catawba, Joseph M.Farley, and Donald C. Cook Nuclear Power Plants for application of the Interim Tube Plugging Criteria (IPC), and the Safety Evaluation issued to Palo Verde Unit 2 prior to restart following the March 14, 1993, tube rupture event.

This emergency change could not be avoided because CECo could not predict the extensive number of tubes with confirmed ODSCC indications which would require plugging/repair. Pre-outage estimates, predicted that approximately 800 tubes would need plugging/repair due to ODSCC. Eddy current inspection of the Braidwood Unit 1 steam generators during Braidwood Unit 1 Cycle 4 Refuel Outage (A1R04) resulted in the identification of 2733 indications at the tube support plates. Of these indications, 1566 were confirmed using RPC inspection techniques. Final inspection results indicated that 1390 tubes would require plugging/repairing due to ODSCC indications at the tube support plate areas based on the present greater than 40% through wall criteria. Application of the 1 volt IPC would allow those tubes which meet the criteria to remain inservice, thus requiring only 426 tubes to be plugged/repaired.

The need for an emergency request was not created by a failure to make a timely application of the Technical Specification Amendment. CECo had no reason to anticipate the extensive number of tubes that would require plugging/repairing at the tube support plate due to ODSCC under the current Technical Specifications. Inspection of the steam generator tubes began on March 21, 1994. When the actual inspection results indicated that the number of tubes needing repairs exceeded the projections, Braidwood considered the options of tube plugging and sleeving all indications. Both options were determined to be unacceptable; tube plugging would result in a significant derating of Unit 1. Sleeving would add approximately 60 days of unplanned outage time. Once decided that the approval of the IPC for Unit 1 was a reasonable option, Braidwood immediately contacted the NRC staff and began preparation of a Technical Specification amendment. Several alternate IPC analytical techniques had to be explored with the NRC before this Technical Specification amendment could be submitted. This Technical Specification amendment has been pursued aggressively and in a timely manner in full consultation with the NRC staff. (Attachment A details the course of events which led to the submittal of this emergency amendment.)

This request for an emergency amendment is necessary to resume power operation for Braidwood Unit 1. Braidwood Unit 1 is currently in Mode 5. Based on the current outage schedule, Braidwood Unit 1 is predicted to be ready, except for approval of this proposed Technical Specification request, to enter Mode 4 on Wednesday, April 27, 1994. Currently, CECo has 3 large units in forced outages requiring major repairs to their main power transformers. This includes Braidwood 2, Powerton 6 and Zion 1. In addition, CECo has 6 other units down for overhauls and also plans on shutting down additional units for repairs prior to June 1. In the event that Braidwood 1 is unable to return-to-service in a timely manner, the additional overhauls may be delayed into June. As a result of these delays, Commonwealth Edison may have inadequate owned reserves as we enter the peak usage period of the year. Non-CECo replacement power, also may not be available because other generators may also be in their peak usage period.

In support of this request, the following information is attached:

Attachment A:	Justification for Emergency Request
Attachment B:	Detailed Description Of The Proposed Changes
Attachment C:	Revised Technical Specification Pages
Attachment D:	Evaluation of Significant Hazards Considerations
Attachment E:	Environmental Assessment
Attachment F:	Westinghouse Letter NSD-TAP-3069, "Braidwood 1: Technical Support for Cycle 5 S/G Interim Plugging Criteria, Pre-WCAP Release," date April 21, 1994
Attachment G:	References

Additional Information in Support of IPC

Included in Attachment B, Braidwood addresses the Draft NUREG-1477 report recommendation that the licensee provide information on various programs, procedures and policies to allow a thorough review of the IPC request. In Attachment B, Braidwood addresses modifications to the current programs which warrants highlighting:

The current Technical Specifications requires a unit shut down if steam generator leakage limit exceeds 500 gpd through any one steam generator. In November 1993, this limit was administratively changed to either a 150 gpd leakrate or a leakrate increase of 25 gpd in 1 hour. This emergency amendment request incorporates the 150 gpd into the Technical Specification. In addition to the reduction in allowable leakage, the Main Steam Line and Steam Jet Air Ejector radiation monitors setpoints have been reduced. Alarm levels for the Main Steam Line radiation monitors would easily be able to detect the 150 gpd leak rate, while the Steam Jet Air Ejector monitors are able to detect leakage below 50 gpd.

Also, the chemistry sampling program has been enhanced to address primary to secondary leakage. The previous chemistry program called for sampling every 72 hours; every 24 if leakage is present; and every 4 hours if leakage is increasing. Changes have been made to increase sampling, if a leak is present, to once every hour or as specified by the Shift Engineer until leakage stabilizes. Operations will review the radiation monitors at least hourly if leakage is detected. More frequent samples will be taken if the radiation monitors show increased levels.

Changes to the leak rate limits and the chemistry sample program have been included in a revision to station procedures for steam generator tube leaks. This procedure provides actions for Operations to mitigate a range of steam generator tube leaks up to those that can be maintained with normal chemical and volume control system lineup. Additionally, the procedure provides two basic methods for leak rate determination to the operators.

Operator training is currently conducted on steam generator tube leaks and ruptures. Additional training will be provided in this area.

IPC Analysis and Technical Support-Proprietary Information

Attachment F provides information in support of the implementation of a proven bobbin coil voltage-based interim plugging criterion. This report contains information which is proprietary to Westinghouse Electric Corporation. Accordingly, we request that this information be withheld from public disclosure. We will comply with the requirements of 10CFR 2.790 to provide proprietary and non-proprietary versions of the material together with an affidavit as soon as the proprietary and non-proprietary versions have been prepared. We will submit the total required number of copies of the proprietary and non-proprietary versions of the information and the required affidavit at that time.

Additional Analysis and Documentation to Follow

In addition to the technical information provided in the attachments, CECo will submit to the NRC the following documents in a timely manner. We are unable to provide this information now because these items warrant additional testing, analysis and review. We are confident that the results of this technical information will strengthen our submittal, and in no way adversely impact safety significance.

- Tube Pull Analysis: Four tubes (3 containing 3 intersections and 1 containing 4 intersections for a total of 13 intersections) were pulled from the Braidwood Unit 1 steam generators during A1R04. These tubes contained some of the largest indications and the largest indication growth rates observed during Cycle 4. CECo is developing a program to analyze these tubes to achieve the following objectives.
  a) Gain additional insight into the root cause of degradation of the Braidwood Unit 1 Steam Generator tubing.
  b) Determine the free span leak rate at main steam line break conditions of the larger indications.
  c) Determine the burst capability of the larger indications.
  d) Assure that the microstructure of the degradation conforms to that required to apply alternate repair criteria for ODSCC in the support plates.

- The Braidwood end of cycle 4 eddy current results will be used to improve the industry's understanding of the probability of detection of larger voltage indications. It is our intent that these data will be used to demonstrate a correlation for ODSCC indications of voltage vs probability of detection.

- The final WCAP-14046 "Braidwood Unit 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria" will be published and submitted to the NRC approximately 4 weeks from the date of this letter.

- A final WCAP will be prepared and transmitted for the Model D4 Steam Generator Tube Support Plate Limited Displacement Analyses within 2-3 months from the date of this letter.

Summary of Results

The probability of a steam generator tube rupture during normal operation is unchanged with the implementation of the IPC proposed for Braidwood Unit 1. As a result of a few large indications observed at end of cycle 4, two additional analysis methodologies have been pursued:

- Evaluation of the limited TSP displacement during a main steamline break event (MSLB).
- Evaluation of a high POD for indications greater than 3 volts.

These evaluations were completed because of the overly conservative assumption of a POD of 0.6 which resulted in an unrealistic overestimation of tube burst probability during the MSLB event.

With limited TSP displacement, the probability of burst during an MSLB is reduced to near zero. Assuming a variable POD, the freespan burst probability is acceptably low to permit full Cycle 5 operation. Furthermore, core damage frequency analysis shows that the contribution to core damage risk would not be significant if any of the above methodologies were applied.

Although CECo believes that Braidwood Unit 1 can be safely operated for the full duration of Cycle 5, we do recognize that the above evaluations and associated assumptions have been the subject of considerable discussion. As such, CECo commits to providing further documented analyses as presented in the previous section and will engage in discussions with the Staff on the appropriateness of continued use of IPC for Cycle 5. Furthermore, if by January 15, 1995, CECo and the Staff cannot agree that continued operation of Unit 1 is appropriate, then Braidwood Unit 1 will be shut down to perform mid-cycle steam generator inspections. If at any time Braidwood has an indication that continued operation at the IPC threshold is unjustified, Unit 1 will be shut down and inspected.

This request for an Emergency Technical Specification Amendment has been reviewed and approved by onsite and off-site review in accordance with Braidwood procedures.

Pursuant to 10CFR50.91(b)(1) a copy of this request has been forwarded to the designated State of Illinois Official.

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respects these statements are not based on my personal knowledge, but on information furnished by other CECo employees, contractor employees, and/or consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

Please address any comments or questions regarding this matter to this office.

Respectfully,

"OFFICIAL SEAL"
JACQUELINE A. SERENA
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 5/14/97

Jacqueline A. Serena

Denise M. Saccomando

Denise M. Saccomando
Nuclear Licensing Administrator

Attachments

cc: R. R. Assa, Braidwood Project Manager - NRR
S. G. Dupont, Senior Resident Inspector - Braidwood
J. B. Martin, Regional Administrator - Region III
Office of Nuclear Facility Safety - IDNS

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ATTACHMENT A

JUSTIFICATION OF EMERGENCY REQUEST

During the most recent steam generator tube inservice inspection for Braidwood Unit 1 a large number of indications were found. Prior to contacting the NRC, Braidwood senior management considered various repair strategies that could be performed within the current Braidwood licensing basis. These strategies were determined to be unacceptable due to either extremely long outage extensions or operational limitations following plant restart. Braidwood initiated conversations with the NRC, and subsequently, decided to apply a previously NRC approved IPC methodology to the inspection results. It quickly became apparent that the previously approved IPC methodology could not reach a successful analytical conclusion to allow the restart of Braidwood Unit 1. An almost continuous dialogue was established between Braidwood and the NRC to discuss deviations from the previously approved IPC methodology. All of the deviations discussed had been previously presented to the NRC in generic industry efforts to develop alternate repair criteria for steam generator tubing. The analysis required to support discussions of these deviations from the previously approved methodology was time consuming. After much discussion, Braidwood chose to adopt the methodology contained in this submittal. It is critical to CECo that this request be processed on an emergency basis to allow Braidwood Unit 1 to restart prior to CECo's peak load season.

The following is a chronological summary of the actions taken by Braidwood to resolve the issue of the large number of indications found during the most recent Unit 1 steam generator tube inservice inspection:

Braidwood Unit 1 entered A1R04 on March 4, 1994. The inspection of the steam generator tubes commenced on March 21, 1994. This inspection, which was originally scheduled to start on March 14, 1994, was delayed because of difficulties completing the containment integrated leak rate test. Braidwood proceeded to perform 100% full length bobbin coil testing on the four steam generators. Data was collected and analyzed. The results identified 2733 indications at the tube support plates. Braidwood conducted inspections of all of these indications with the use of a RPC. By March 26, 1994, Braidwood recognized that the pre-outage estimate of the number of tubes requiring repair using the existing criteria for actions in response to ODSCC was too low. New estimates of approximately 1500 tubes requiring repair were derived from the initial bobbin coil results and the percentage of those indications that were confirmed by RPC inspections.

Braidwood considered various alternatives to declare the steam generators operable per the current Technical Specifications. Sleeving or plugging of the tubes would result in:

- reduced Unit 1 power output
- reduced reactor coolant flow margin,
- increase in unplanned personnel exposure,
- increase in unplanned refueling outage time, and
- increase in overall outage cost.

Additionally, installation of steam generator tube sleeves now, may preclude the use of direct tube repair in the future, if such technology becomes available.

Braidwood considered the option of applying for an amendment which allowed for the use of the Interim Plugging Criteria (IPC). Commonwealth Edison was fully aware of the NRC and industry's efforts in development of the industry Alternate Repair Criteria (ARC). Braidwood Station fully anticipated requesting a Technical Specification amendment this fall after the issuance of the anticipated ARC Generic Letter. Braidwood decided that it was important to contact the NRC to apprise them of the current inspections results and discuss the possibility of amending the Technical Specification for the use of IPC for Unit 1 during A1R04. The following describes the history of communications which took place between CECo and the NRC prior to the submittal of this emergency request.

> On March 28, 1994, Braidwood Station updated the NRC on their inspection results and the results expected from the remaining inspections. At that time, 42% of the tubes had been inspected and it was predicted that up to 1500 tubes would require plugging/repairing. CECo informed the NRC that they were considering asking for an IPC amendment in accordance with the anticipated Generic Letter. The NRC staff informed Braidwood that a 1 volt limit would have to be applied because Braidwood's generators had 3/4" diameter tubes. Braidwood discussed briefly the use of tube expansion in the tube support plate region to further limit the probability of a tube rupture and tube support plate displacement. The staff responded that it would be unable to approve the use of tube expansion in the analysis in the short time that remained before the amendment request was needed. Discussions with the Staff and CECo focused on previously approved Safety Evaluation Reports issued for Catawba, Farley and Cook and Draft NUREG-1477 as a bases for this request. Braidwood informed the staff that a 10.4 volt indication was found in the "D" generator. Braidwood and the Staff discussed the growth rate of the indications, plans for tube pull, and the potential impact that such growth rate would have on the uapplication of the previously approved IPC amendments.

On March 30, 1994, the NRC, Braidwood and Westinghouse held a follow-up teleconference to discuss the latest results. Braidwood stated that 60 to 70% of the steam generator tubes had been analyzed. Using the voltage indications that were detected and the analysis methods described in other IPC applications, Braidwood calculated that the probability of burst would be unacceptable using some NUREG 1477 mandated analysis requirements. Braidwood then discussed the possibility of using a probability of detection (POD) based upon voltage in lieu of the standard 0.6 which is part of the draft NUREG. Braidwood also discussed the use of the tube support plate limited displacement analysis in the submittal. A similar analysis had been previously submitted for Catawba and was being reviewed by Pacific Northwest Laboratories (PNL). The NRC felt that use of a voltage dependent POD could not be reviewed in a timely manner. The NRC agreed to provide feedback on: other previous industry submittals, and the tube support plate limited displacement analysis that was being reviewed by PNL.

On April 4, 1994, a conference call was held between Braidwood, the NRC, PNL and Westinghouse. PNL and Westinghouse discussed information that was pertinent to the tube support plate limited displacement analysis. The differences between the Braidwood D4 steam generator tube support plate and the Catawba model D3 generator plates were explained. The NRC said that they would continue their discussions with PNL and would be able to provide CECo with some feedback on the use of this analysis during the next conference call. Braidwood explained their overall analysis plan: tube expansion in conjunction with reliance upon limited tube support plate displacement. An agenda was discussed for a meeting at NRR to further discuss inspection results and the pending request. This meeting was scheduled for April 8, 1994.

During an April 6, 1994, teleconference the staff informed CECo that it would further consider the use of the tube support plate limited displacement analysis. Braidwood discussed the tubes that were being considered for tube pull. The Staff stated that it would not be able to comment on the value or future value of tube expansion at that time.

On April 8, 1994, a meeting was held at NRR with the NRC, CECo and Westinghouse. Braidwood discussed the current outage schedule. Steam generator tube inspection scope and results were also discussed. A total of 1390 tubes would require plugging if the current Technical Specification acceptance criteria was applied. If a 1 volt IPC was applied, then 426 tubes would require plugging. The IPC implementation plan was discussed, including the tubes which were planned to be pulled. Westinghouse then discussed: overall IPC and tube integrity approach, the tube support plate limited displacement analysis and tube expansion process. It was agreed that credit would not be taken for the tube expansion process and that the tube support plate limited displacement analysis would be used for IPC justification. The Staff agreed to provide some feedback to Braidwood.

On April 11, 1994, a followup call was held. The Staff indicated that Braidwood should not perform tube expansion at this time. They noted that there could be metallurgical and long term corrosion concerns at the expanded intersections. CECo decided that it would not perform tube expansion during A1RO4. CECo discussed that the current accident analysis estimate for steam generator leakage using the log-logistic approach would be approximately 34 gpm. This estimated leak rate would have to be significantly reduced to meet the Standard Review Plan acceptance of 4.6 gpm. CECo committed to having additional leak rate calculation done for the next conference call.

A followup call was held on April 13, 1994. The NRC was given additional data on steam line break leak rate analyses. This data illustrated that use of the Draft NUREG-1477 would result in estimates as large as 89.7 gpm while use of the industry leakage model predicted 4.2 gpm. CECo concluded that the steam line break accident leakage analysis portion of the submittal would need to incorporate the industry leakrate model that had not been previously approved by the Staff. Also, during this call, Braidwood stated that the steam line break analysis would be based upon using a POD of 0.6 and the limited tube support plate displacement. The Staff agreed to review Braidwood's submittal and stated that the site needed to submit the Technical Specification amendment and the various sections for the technical justification as soon as possible. Braidwood began providing the various sections of the submittal on April 15, 1994.

On April 20, 1994, the Staff contacted Braidwood and said that they would not be able to approve the tube support plate limited displacement analysis in support of Braidwood's proposed amendment to the Technical Specifications in a timely manner for this outage. They suggested that use of a voltage based POD could be an option that CECo would want to include in the submittal.

On April 21, 1994, a followup phone was held between the Staff and CECo. The discussion focused on CECo's amendment request and the applicability of a risk based assessment analysis. Palo Verde previously submitted this type of analyses for the restart of Unit 2 after their tube rupture. CECo agreed to review Palo Verde's analysis and to submit in the amendment request a section on risk based assessment.

Since March 26, 1994, when Braidwood along with Westinghouse recognized that the pre-outage estimates of tubes requiring plugging/repair were too low, Braidwood has taken aggressive continual actions to resolve the issue of steam generator operability. Because of the larger than anticipated growth rates, previously approved IPC methodology could not be simply applied. The review of various alternate analytical techniques with the NRC was conducted on an appropriately aggressive schedule. This, along with the complexity of the analyses precluded the amendment request from being submitted earlier.

In addition to Braidwood's inability to anticipate that this amendment request would be necessary on an expedited bases, this request for an emergency amendment is necessary to resume power operation of Braidwood Unit 1 in a timely manner. Currently, CECo has several units in forced outages. The return-to-service for Braidwood Unit 1 in a timely manner may be critical to CECo's ability to meet our load requirements for May. As a result of any additional delays, Commonwealth Edison may have inadequate owned reserves as we enter the peak usage period of the year.

ATTACHMENT B

DESCRIPTION AND SAFETY ANALYSIS OF PROPOSED CHANGES TO APPENDIX A TECHNICAL SPECIFICATIONS OF FACILITY OPERATING LICENSE NPF-72

A. <u>DESCRIPTION OF THE PROPOSED CHANGE</u>

Commonwealth Edison Company (CECo) proposes to amend the following Technical Specifications:

Specification 3.4.5	REACTOR COOLANT SYSTEM-STEAM GENERATORS
Specification 3.4.6.2	REACTOR COOLANT SYSTEM-OPERATIONAL LEAKAGE

This proposed license amendment request will modify Specification 3.4.5 to allow an eddy current bobbin coil probe voltage based steam generator (SG) tube support plate (TSP) interim plugging criteria (IPC) to be applied for Braidwood Unit 1 Cycle 5.

This proposed license amendment request will also modify Specification 3.4.6.2 to reduce the allowable reactor-to-secondary leakage from 1 gallon per minute (gpm) total through all SGs and 500 gallons per day (gpd) through any one SG to 600 gpd total through all SGs and 150 gpd through any one SG.

Technical Specification Bases Sections 3/4.4.5, STEAM GENERATORS, and 3/4.4.6.2, OPERATIONAL LEAKAGE, will also be modified to reflect these changes, respectively.

B. <u>DESCRIPTION OF THE CURRENT REQUIREMENT</u>

<u>Specification 3.4.5</u>

The Technical Specification Surveillance Requirements (TSSRs) associated with Specification 3.4.5 currently require that any SG tube found during the SG tube inservice inspection with an indication exceeding the plugging or repair limit of 40% of the nominal wall thickness be removed from service by plugging or repaired by sleeving.

Specification 3.4.6.2

Specification 3.4.6.2.c currently requires that reactor coolant system (RCS) leakage shall be limited to 1 gpm total reactor-to-secondary leakage through all SGs not isolated from the RCS and 500 gpd through any one SG.

C. BASES OF THE CURRENT REQUIREMENT

Specification 3.4.5

The TSSRs for inspection of the SG tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of SG tubes is based on a modification of Regulatory Guide (RG) 1.83, "Inservice Inspection of PWR Steam Generator Tubes," Revision 1, July 1975. Inservice inspection of SG tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of SG tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

Specification 3.4.6.2

The total SG tube leakage limit of 1 gpm for all SG not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Title 10, Code of Federal Regulations, Part 100 (10 CFR 100) dose guideline values in the event of either a SG tube rupture or a main steam line break (MSLB). The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per SG ensures that SG tube integrity is maintained in the event of a MSLB or under loss of coolant accident (LOCA) conditions.

D. NEED FOR REVISION OF THE REQUIREMENT

During the Braidwood Unit 1 Cycle 4 Refuel Outage (A1R04) which began March 4, 1994, a SG tube inservice inspection was performed in accordance with TSSR 4.4.5.0. The results of this inspection indicated that under the current technical specification acceptance criteria a total of 1423 SG tubes, of which 1390 are due to ODSCC at the TSPs, would have to be removed from service by plugging or repaired by sleeving. Additionally, the distribution of these SG tubes would cause a large disparity in the number of tubes removed from service between SGs "B" and "C." This disparity between SGs "B" and "C" would probably cause a noticeable RCS flow imbalance and result in potential RCS loop power asymmetries. Plugging of all tubes would require re-analysis since SG "C" would exceed the currently analyzed plugging limit. Sleeving of even the minimum number of tubes necessary in SG "C" to conform with the current analysis would greatly increase the cost of SG repair and result in a significant extension of the outage critical path. This option would also limit the unit to approximately 90% of rated thermal power.

E. DESCRIPTION OF THE REQUESTED REVISION

Specification 3.4.5

Technical Specification Surveillance Requirement (TSSR) 4.4.5.2, Steam Generator Tube Sample Selection and Inspection Requirements, will be changed to require a 100% bobbin coil probe inspection for all hot leg tube support plate intersections and all cold leg intersections down to the lowest cold leg tube support plate with outside diameter stress corrosion cracking (ODSCC) indications. An inspection using a rotating pancake coil (RPC) probe is required in order to show operability of tubes with flaw-like bobbin coil signal amplitudes greater than 1.0 volt but less than or equal to 2.7 volts. For tubes that will be administratively plugged or repaired, no RPC inspection is required. The RPC results are to be evaluated to establish that the principle indications can be characterized as ODSCC.

TSSR 4.4.5.4.a.6, Plugging or Repair Limit, will be changed to specify that this definition does not apply for Unit 1 Cycle 5 to the region of the tube subject to the TSP IPC limit, i.e. the TSP intersections, and that Specification 4.4.5.4.a.11 describes the repair limit for use within the TSP intersection of the tube.

TSSR 4.4.5.4.a.11, Tube Support Plate Interim Plugging Criteria Limit, will be added to define the TSP IPC limit. The TSP IPC limit is used for the disposition of a SG tube for continued service that is experiencing ODSCC confined within the thickness of the TSPs. For application of the TSP IPC limit, the tube's disposition for continued service will be based upon standard bobbin coil probe signal amplitude of flaw-like indications. The plant specific guidelines used for all inspections shall be amended, as appropriate, with respect to the voltage/depth parameters specified in Specification 4.4.5.2. Pending incorporation of the voltage verification requirements in American Society of Mechanical Engineers (ASME) standard verifications, an ASME standard calibrated against the laboratory standard will be utilized in Unit 1 steam generator inspections for consistent voltage normalization.

1. A tube can remain in service with a flaw-like bobbin coil signal amplitude of less than or equal to 1.0 volt, regardless of the depth of the tube wall penetration, if, as a result, the projected end of cycle distribution of crack indications is verified to result in total primary to secondary leakage less than 9.4 gpm (includes operational and accident leakage). The basis for determining expected leak rates from the projected crack distribution is provided in Westinghouse letter report NSD-TAP-3069, "Braidwood 1: Technical Support for Cycle 5 S/G Interim Plugging Criteria, Pre-WCAP Release," dated April 21, 1994 (NSD-TAP-3069).

2. A tube can remain in service with a flaw-like bobbin coil signal amplitude greater than 1.0 volt but less than or equal to 2.7 volts provided an RPC inspection does not detect degradation.

3. A tube with a flaw-like bobbin coil signal amplitude of greater than 2.7 volts shall be plugged or repaired.

Certain tubes identified in NSD-TAP-3069, shall be excluded from application of the TSP IPC limit. It has been determined that these tubes may collapse or deform following a postulated LOCA coincident with a safe shutdown earthquake (SSE).

TSSR 4.4.5.5, Reports, will be changed to add a requirement to report to the Nuclear Regulatory Commission (NRC) all tubes on which the TSP IPC has been applied. This addition also delineates what information must be included in the report.

Specification 3.4.6.2

Specification 3.4.6.2.c will be changed to reduce reactor-to-secondary leakage from 1 gpm to 600 gpd total through all SGs and from 500 gpd to 150 gpd through any one SG.

Technical Specification Bases Sections 3/4.4.5, STEAM GENERATORS, and 3/4.4.6.2, OPERATIONAL LEAKAGE, will also be modified to reflect these changes, respectively.

The specific changes to these technical specifications and associated bases are included in Attachment C.

F. BASES FOR THE REVISED REQUIREMENT

The basis for Braidwood's request is:

1. The approval of similar requests for IPCs for other plants with 3/4 inch and 7/8 inch diameter SG tubing.

2. The NRC's ongoing review of Electric Power Research Institute (EPRI) Draft Report TR-100407, "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates," Revision 1, August 1993, and EPRI Draft Report NP-6864-L, "PWR Steam Generator Tube Repair Limits: Technical Support Document for Expansion Zone PWSCC in Roll Transitions - Rev. 2," Revision 2, August 1993.

3. An understanding of the NRC and industry desire to improve the basis for steam generator tube repair.

4. The completion of a satisfactory review assuring the structural integrity of Braidwood SG tubing during continued operation.

To support this request for amendment, Braidwood has removed four tubes for laboratory examination, leak, and burst testing (Refer to Table 2). These tubes were removed based on their indication size, having two or three support plate intersections intact as well as the flow distribution baffle intersection. In addition, analyses for probability of burst and end-of-cycle (EOC) primary-to-secondary leakage under MSLB conditions has been completed.

APPROACH

The methodology for developing the Braidwood IPC is described below:

1. Calculate the allowable primary to secondary leakage during a MSLB event based on a small fraction of 10 CFR 100 limits at the site boundary.

2. Develop an in-service voltage distribution using the Draft NUREG-1477, "Voltage Based Interim Plugging for Steam Generator Tubes - Task Group Report" Probability of Detection (POD) of 0.6.

3. Evaluate Cycle 4 growth rates.

4. Predict EOC voltage distribution.

5. Evaluate EOC voltage distribution for tube burst and leakage during a MSLB event.

6. Use the limited support plate displacement evaluation for Model D-4 SG to limit the potential for tube burst during cycle 5.

7. Apply the industry developed EPRI leak rate correlation using data and correlation techniques in Reference 4.

8. Apply a voltage dependent ODSCC POD.

9. Calculate an anticipated leak rate during MSLB.

10. Compare the calculated leak rate with the site allowable leak rate calculated in step 1.

11. Declare the SG operable if the anticipated EOC leak rate is less than the site allowable leak rate.

12. Assess the change in core damage frequency assuming a MSLB with the probabilty of burst as evaluated both BOC 5 and EOC 5.

This approach ensures the safe and reliable start up and operation for Braidwood Unit 1 Cycle 5 using the proposed 1.0 volt IPC.

The results of supplemental analyses, including examination and testing of the four tubes removed from the steam generators and the final Model D-4 limited tube support plate displacement analysis will be completed in an expedient manner.

LEAK RATE MODEL

An important element in the Braidwood Unit 1 analysis is the use of the industry developed leak rate correlation for primary to secondary leakage through cracked tubes at MSLB differential pressures. The development of this correlation included analyzing tubes pulled from operating plants and tubes with prototypic ODSCC developed in model boilers.

A number of criteria have been established for the exclusion of points from the data base based upon non-prototypic degradation, test errors, poor test conditions, or damage during tube removal from the SGs. These criteria have recently been reviewed by the Ad-Hoc Committee for Alternate Repair Limits,and were used for development and/or exclusion of data from the correlation in the Braidwood Unit 1 analysis.

STEAM GENERATOR TUBE INTEGRITY

The purpose of the Technical Specification repair limit is to ensure that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operation, transient and postulated accident conditions. The proposed repair limit for Braidwood Unit 1 involves the implementation of the voltage based eddy current signal amplitude of 1.0 volt for ODSCC occurring within the thickness of the tube support plates. This is a change from the traditional 40% depth based criteria which is currently required by Braidwood Technical Specifications. The IPC proposed for Braidwood Unit 1 meets the requirements of RG 1.121 by demonstrating that tube leakage is acceptably low and tube burst is a highly improbable event during normal operation as well as a MSLB event.

EDDY CURRENT EXAMINATION

Extensive eddy current examination of SG tubing has been the characteristic of Braidwood Unit 1 refueling outages. The standard has been to fully understand the SG conditions prior to beginning a new fuel cycle. Unit 1 has had 100% of its tubes inspected at each refuel outage since plant startup. Each inspection has been conducted using the most up-to-date technology available. Enhancements to the bobbin coil inspection program have included:

1. routine random RPC inspections of known degraded intersections,

2. application of 3 coil RPC to determine degradation orientation,

3. the use of probe wear standards to assure reliable bobbin coil test data,

4. a requirement for the use of Qualified Data Analysts examined under the industry EPRI performance demonstration program.

To assure tube integrity, eddy current exams were conducted on all tubes in the Braidwood Unit 1 SG during A1R04. Eddy current data collection, data analysis, and probe change out requirements were used for the full inspections in anticipation of potential future implementation of alternate repair criteria. ASME calibration standards were cross calibrated to reference IPC laboratory standards. Eddy current probes were changed out when the probe wear standard indicated greater than or equal to 15% deviation from the initial calibration using a four 100% through wall hole standard. This resulted in greater reproducibility of results and reduced noise due to probe wobble and centering effects.

Data analysis was completed by analysts specifically trained to evaluate ODSCC defect sizing. In addition, 22 of the 42 (52%) analysts were trained and qualified to the industry standard Qualified Data Analysis program.

To evaluate Cycle 4 growth rates as indications of ODSCC at EOC 4, the EOC 3 inspection results were reevaluated to ensure accurate Cycle 4 growth analysis. Braidwood Unit 1 had the advantage of a 100% inspection at the EOC 3 as well as a 100% inspection of SG C during October 1993 as the result of a primary to secondary tube leak unrelated to tube support plate ODSCC. This allowed ODSCC growth evaluations for SG "C" for the periods October 1992 to October 1993, November 1993 to March 1994, and a growth evaluation of SGs "A," "B," and "D" for the entire Cycle 4. The results of this growth rate analysis were conservatively applied to the indications remaining in service. The average growth for the full Cycle 4 in SGs "A," "B," "C," and "D" was 0.23 volts per effective full power year (EFPY) with the average growth in the first portion of Cycle 4 in SG "C" being 0.19 volts per EFPY. During the last portion of Cycle 3, the average growth rate in SG "C" was 0.11 volts per EFPY.

A few indications showed abnormally large growth rates during Cycle 4, with the largest growth being 9.76 volts. Therefore, Braidwood chose to perform additional analyses to demonstrate that limited SG tube support plate displacement during MSLB events will result in less than 0.5 inch of tube support plate motion for a large fraction of the tubes. This analysis shows that limited support plate motion will not allow a crack contained within the support plate to become completely uncovered and thus will not allow excessive leakage during a MSLB event for a large fraction of the tubes. For conservatism, the MSLB leak rates are calculated assuming freespan or completely uncovered tubes. Limited plate movement is applied only to assess burst margins.

The leakage assessment during a MSLB event on the worst SG results in a maximum anticipated leak rate of 3.0 gpm. This is more than a factor of 3 lower than the allowable 9.1 gpm primary to secondary leak rate limit with containment bypass during a MSLB. These two evaluations for leakage and burst probability result in the conclusion that Braidwood Unit 1 SGs retain adequate structural integrity as defined in RG 1.121.

PROBABILITY OF DETECTION

The POD of indications is an important consideration for the development and implementation of interim plugging criteria and alternate repair criteria. This issue has been the subject of much discussion between the industry and NRC as part of the joint NRC and Industry Technical Steering Committees for steam generator rulemaking. The analysis contained in NSD-TAP-3069 is completely based upon two assumptions:

(1) Draft NUREG-1477 comment resolution that POD is fixed at 0.6

(2) Limited support plate motion provides the necessary margin to burst for both beginning-of-cycle (BOC) 5 and EOC 5.

It has been recognized from the beginning of the development of the Braidwood Unit 1 request for IPC that a voltage dependent POD for ODSCC indications in Braidwood would be a more appropriate methodology for assessing both beginning of cycle and end of cycle burst probability. This was driven by the presence of a 10.44 volt indication at EOC 4 in Braidwood Unit 1.

Based upon data collected as part of the EPRI Performance Demonstration Program (Refer to Figure 1) illustrates the performance of some 20 eddy current data analysts evaluating data from a unit with 3/4" inside diameter and 0.049" wall thickness tubes. This figure clearly demonstrates the voltage dependence of the POD and argues for a POD > 0.6 for ODSCC indications larger than 1.0 volt.

As part of the growth rate data analysis, a re-analysis of Braidwood 1992 EOC 3 eddy current data for the 2733 indications was completed. The objective of this analysis was to find and size all possible indications. This sizing was done using extremely conservative techniques developed to identify and size possible indications. The result could possibly over estimate the size of indications not identified at EOC 3. It is important to remember that all indications distorted or sized during the EOC 3 inspection had RPC data collected. Only those which had no indication of crack like signals with the RPC were left in service. All indications of crack like structure by RPC were plugged at EOC 3.

An analysis of this re-sized EOC 3 data shows that the largest indication that could have been left in service EOC 3 was less than 2 volts. In fact, 17 indications greater than 2 volts were identified during EOC 3 and repaired. When the EOC 3 data was re-analyzed no indications greater than 2 volts were identified.

This assessment gives CECo a high level of confidence that indications greater than 2 volts can indeed be identified and the appropriate action taken based upon experience at Braidwood Unit 1. One of the reasons for the possible improvement in detection of indications at Braidwood as compared to other units is the lack of copper in the system. All Braidwood Unit 1 secondary system heat exchanger tubing is stainless steel and all piping is carbon steel, so the absence of copper improves eddy current detection capability.

To support the contention that a very high probability of detection for indications greater than or equal to 3 volts is indeed reality at Braidwood, CECo is evaluating the possibility of a "blind" re-analysis of EOC 3 and EOC 4 data to attempt to define an ODSCC POD curve for a plant without copper in the secondary system. It is recognized that such re-analysis may be criticized relative to the validity of base truth on specific indications and the possibility of bias by participants knowing that ODSCC is indeed a particular form of degradation in the Braidwood SGs. However, the only other known way to assure a high confidence in a POD discussion is to base it exclusively on tubes pulled from service following inspection. To obtain a statistically significant data base would require the pulling of several dozen, if not hundreds, of tubes. Based on CECo's understanding of this mode of degradation, the data accumulated from tube pulls at Catawba and in Europe, and the Braidwood Unit 1 tube pulls, we will slowly improve the POD versus voltage correlation. As more utilities use the ODSCC alternate repair criteria and remove more tubes for destructive evaluation, it will be possible to add additional data to the POD data base, thereby providing improved confidence. Until such a pulled tube database is accumulated reliance on other methodologies for development of POD versus voltage must be used.

This assessment of the probability of detection of ODSCC indications reduces the probability of burst BOC 5 to 2E-3 (assuming a MSLB has occurred) which is acceptably low. The EOC probability of burst using a POD of 1 for indications greater than 3 volts is on the order of 5E-3. These probabilities of burst with free span indications unsupported by the support plate are acceptably low to permit full cycle operation of Braidwood Unit 1 Cycle 5.

RISK EVALUATION OF CORE DAMAGE

As part of CECo's evaluation of the operability of Braidwood Unit 1 Cycle 5, a risk evaluation was completed. The objective of this evaluation was to compare core damage frequency, with containment bypass, with and without the interim plugging criteria applied at Braidwood 1.

CECo has evaluated the impact of operation using the proposed interim plugging criteria against the results of insights from the draft Braidwood Individual Plant Examination (IPE). While the Braidwood IPE is not in its final form, it is believed that the quantification in hand is sufficiently robust to allow a validation assessment of the impact of such operation. The CECo evaluation parallels that described in the NRC Staff's Safety Evaluation Report for Palo Verde Unit 2 dated August 19, 1993.

The values calculated in NSD-TAP-3069, for BOC 5 and EOC 5 using 0.6 POD were used to develop a cycle average burst probability. Another BOC 5 burst probability assuming a POD of 0.6 for indications less that 3 volts and 1.0 for indications greater than 3 volts (Refer to Table 1), was used to evaluate the impact of POD on core damage frequency.

The total Braidwood core damage frequency is estimated to be 2.74E-5 per reactor year with a total contribution from containment bypass sequences of 2.9E-8 per reactor year in the current IPE. Operation with the alternate repair criteria with a variable POD is expected to increase the MSLB with containment bypass sequence frequency contribution by a factor of only 10%. An upper bound increase of a factor of two is derived when the fixed POD of 0.6 is employed in the calculation. Neither increase is significant from a risk perspective.

The reason for a reduced core damage frequency with a higher POD is that large voltage indications have a high assurance of being identified and removed from service during inspection. Therefore, the calculation of burst probability during MSLB changes because of differences in the assumed distribution of indications left in service at BOC. The EOC burst probability also changes because the growth distribution is added to the new BOC distribution of indications. The result of this change, as shown in Table 1, is a significant reduction in burst probability during MSLB.

Therefore, the operation of Braidwood Unit 1 Cycle 5 for a complete 18 month fuel cycle with the application of the one volt IPC does not significantly increase the core damage frequency even with the conservative assumption of a POD of 0.6 and application of the full growth rate distribution observed during Cycle 4.

OTHER CONSIDERATIONS

The following additional administrative or procedural actions are in place or planned in support of this proposed license amendment request.

1. Actions being taken to mitigate the corrosive environment in the TSP crevices and to ensure that future growth rates and crack morphologies will be within expected bounds.

    Braidwood has implemented the following actions;

    Industry Guidelines

    -- Full compliance with all EPRI Secondary Chemistry Guidelines, Revision 3.

    SG Tubing Crevice Fouling

    -- Maintaining hotwell dissolved oxygen (DO) concentrations < 3 parts per billion (ppb).

    -- Use of advanced amines such as ethanolamine (ETA) for secondary pH control.

SG Crevice pH

-- With the start of Braidwood Unit 1 Cycle 5 will begin the addition of boric acid to the secondary side for mitigation of SG TSP ODSCC per EPRI boric acid applications guidelines. Both low power boric acid soaks during startup and full power operation on boric acid chemistry will be implemented.

-- Performance of SG hideout return studies during shutdown to assess the impact of operating chemistry on SG crevice chemistry and potential formation of caustic crevices which can cause TSP ODSCC.

SG Sodium Reduction

-- The molar ratio control program begun at the start of Cycle 4 will be maintained. This program adjusts the sodium to chloride ratio in the steam generators by adding ammonium chloride to the condensate system.

-- Installed a reverse osmosis unit in the makeup water system to reduce overall impurity input to the secondary side, including sodium.

-- Performed 100% eddy current inspection of 3 out of the 4 condenser water boxes over the last two refuel outages.

SG Electrochemical Potential (ECP) Reduction

-- Continued use of high hydrazine concentrations for maintaining reducing conditions in the SGs and passivation of piping systems and components.

-- Braidwood does not have any copper components in its feedwater or condensate systems.

Currently, Braidwood, in conjunction with our Corporate Chemistry Department, is pursuing several methods to further enhance our SG corrosion control program. They are as follows:

-- Secondary side SG cleaning during the Braidwood Unit 1 Cycle 5 Refuel Outage (A1R05), either in the form of chemical cleaning or pressure pulse cleaning.

-- The addition of other amines either with ETA or in place of ETA will be evaluated to optimize the pH control and minimize iron transport.

-- Chemistries are being evaluated to improve iron transport out of the SGs. The goal is to increase the efficiency of iron removal via the blowdown system.

2. Description of the site-specific leak rate monitoring procedures and an assessment of their effectiveness for ensuring timely detection, trending, and response to rapidly increasing leaks.

Background

In October 1993, Braidwood Unit 1 experienced a primary to secondary steam generator tube leak on the order of 300 gpd. The unit was shutdown well below the Technical Specification limit of 500 gpd. All chemistry and radiation monitoring equipment performed as expected. The leaking tube was found to be located in the U-Bend portion in the "C" steam generator. As a result of this event Braidwood has implemented program enhancements as described in the following.

Ability to Detect Leakage

Braidwood uses a combination of radiation monitors and central processing unit (RM-11) to monitor for primary to secondary leakage. During the event of October 1993 operations used the RM-11 system to trend the main steam line and the steam jet air ejector radiation monitors.

The main steam line radiation monitors properly identified the leaking SG and alarmed at the alert setpoint. Unit operations was notified of increase in activity via the RM-11 system alarming in the control room. With a low RCS radioactivity, the monitors showed a clear increase in secondary activity. The alert setpoints were reached 17 and 13 minutes after the initial radioactivity increase on the main steam line and steam jet air ejector radiation monitors, respectively. The main steam line radiation monitors (Unit 1 and Unit 2) have had their alert setpoint lowered from 0.25 mr/hr to 0.2 mr/hr. This corresponds to a primary to secondary leak of about 150 gpd. The alarm setpoint has also been lowered from 0.5 mr/hr to 0.4 mr/hr.

The steam jet air ejector radiation monitor also exceeded its alarm setpoint in this event. It was decided to use the experience of the October 1993 event to rescale the monitor. The new alarm setpoint was based on linear scaling of the reading seen at 300 gpd to the expected reading at 150 gpd. The steam jet air ejector radiation monitors (Unit 1 and Unit 2) have had their alarm setpoint lowered from 1.15E-5 micro curies/cc to 1.00E-5 micro curies/cc. The alert setpoint has also been lowered from 5.75E-6 micro curies/cc to 3.00E-6 micro curies/cc. This corresponds to a primary to secondary leak of about 50 gpd. This is based on the plant conditions at the time of the October 1993 event. The correlation of micro curies/cc to gpd leakage will vary slightly based on the amount of failed fuel, flowrate of the stream being sampled, and the degassing efficiency of the condenser. To ensure proper leak detection capability the operability of these monitors will be given a priority equivalent to that of the main steam line radiation monitors.

The time to confirm that a leak exists via sampling is adequate (1.5 hours or less). Chemistry has implemented more specific instructions via procedure changes for personnel to perform a "quick count" to ensure confirmation is rapid.

Controls When Known Leakage Exists

The previous chemistry sampling program calls for sampling every 72 hours; every 24 hours if leakage is present; and every 4 hours if leakage is increasing. Changes have been made to the chemistry procedures which state that if leakage has been detected sampling should be performed once per hour until the leakrate is stable and then reduced in frequency not to be less than once per day. If the leakage increases at a rate of greater than or equal to 25 gpd in a 1 hour period, sampling frequency is increased to once every hour or as specified by the Shift Engineer, until leakage stabilizes. Additional monitoring has also been proceduralized in Braidwood Operating Abnormal Procedure (BwOA) SEC-8, STEAM GENERATOR TUBE LEAK. This calls for operations to review radiation monitors hourly if leakage is detected. More frequent samples will be triggered if the radiation monitors show increased levels.

Braidwood is currently operating with an administrative primary to secondary leakage limit of 150 gpd or an increase of 25 gpd in 1 hour. Upon acceptance of this IPC submittal the Braidwood Technical Specification primary to secondary leakage limit will be reduced to 150 gpd.

Operator Training

Tube rupture scenarios are conducted frequently in the simulator. Scenarios include varying radiation monitor responses. Additional training will be given in this area as well as main steam line break scenarios.

Operator Actions for an SG Tube Leak

Abnormal radiation in a steam generator indicates primary to secondary leakage. This can be shown by trends or alarms on main steamline, steam jet air ejector, or SG blowdown radiation monitors, or from chemistry samples. A large leak could be indicated by feedwater flow being less than steam flow, decreasing feed flow, or decreasing feed regulating valve position in conjunction with a stable steam generator level. These symptoms, however, would more likely be noticed with a tube rupture. Any of these indications require entry into BwOA SEC-8. This procedure provides actions to mitigate a range of steam generator tube leaks up to those that can be controlled with a normal chemical and volume control system lineup (approximately 120 gallons per minute). The basic flow of this procedure is as follows:

a. Enter the procedure based on any indication of leakage. This includes an increase in pre-existing leakage (i.e. increasing trends on main steamline or steam jet air ejector rad monitors, increased activity in Chemistry samples, or an RCS charging-letdown flow imbalance.)

b. Take actions to maintain RCS inventory. Trip the reactor and actuate safety injection if inventory can not be maintained.

c. Minimize secondary system contamination.

d. Perform a gross leakrate check to determine if immediate plant shutdown should be commenced.

e. Trend SG leakrate data using rad monitor indications. The Chemistry Department is also contacted at this time to perform leakrate determinations.

f. Determine if shutdown is required based on either absolute leakrate or on a change in leakrate. Exit the procedure if shutdown is not required.

g. Initiate a unit shutdown.

h. Identify the leaking SG. (After this point the actions are very similar to Emergency Procedure actions for a SG tube rupture.)

i. Isolate the leaking SG.

j. Cooldown and depressurize the RCS to stop leakage.

k. Cooldown the leaking SG.

BwOA SEC-8 provides two basic methods of leakrate determination to the operators. Leakrates for large leaks (10 gpm or greater) can be determined by the operator from a charging-letdown flow balance on the RCS. Leakrates for smaller leaks use radiation levels in the secondary system and are determined by the Chemistry Department. Main steamline and steam jet air ejector radiation monitor status is trended in the main control room on the RM-11. Stable radiation monitor indications are indicative of a stable leakrate. For stable radiation monitor readings, shutdown requirements are based on absolute leakrate values. Should radiation monitor readings show a rapid increase during a shutdown based on absolute leakrate values, the allowed time to complete the shutdown is reduced to the shortest time practical for a controlled shutdown (4 hours).

If secondary radiation monitor readings are increasing and the absolute leakrate, then the change in leakrate shutdown requirements are applied. The change in leakrate shutdown requirements are only applied if absolute leakrate exceeds 50 gpd since this is considered the lowest value that can be accurately measured.

Three general sizes of leaks were considered in writing this procedure (see Table 3).

The first was a leak of less than 150 gpd that was changing less than 25 gpd per hour. For this leak size the operator would initiate actions to minimize contamination, monitor releases and leak size, and then exit the procedure. BwOA SEC-8 would be reentered if the leak size were to increase.

The second leak size to be considered was one less than 10 gpm and either 1) greater than 150 gpd, or 2) changing by greater than 25 gpd per hour but less than 100 gpd per hour. For this leak size, the operator would again initiate actions to minimize contamination, monitor releases, and monitor leak size.

The operator would determine the time requirement for shutdown and leave BwOA SEC-8 to perform a shutdown and cooldown using the Braidwood General Procedures (BwGPs). When the unit reached MODE 5 the SG would be isolated for repairs. During the shutdown BwOA SEC-8 would be reentered if the leak size were to increase.

The third size leak was one either larger than 10 gpm, or less than 10 gpm but increasing by greater than 100 gpd per hour. In this case the operator would shutdown the reactor per the BwGPs but remain in BwOA SEC-8 to perform SG tube leak recovery.

A leak size of greater than 120 gpm would require a Reactor trip and Safety Injection and would be addressed as a tube rupture by the Braidwood Emergency Procedures (BwEPs).

<u>Procedural Adequacy</u>

The BwEPs have been reviewed in light of the Institute for Nuclear Power Operations (INPO) Significant Operating Experience Report (SOER) 93-01, "Diagnosis and Mitigation of Reactor Coolant System Leakage Including Steam Generator Tube Ruptures." Braidwood procedural diagnosis is adequate and includes the use of radiation monitors and alternate indications. The diagnosis is treated as a continuous action step. The chemistry procedures for confirming and calculating leakage were also reviewed and found to be adequate.

3. Upgrade guidelines for the eddy current test and data analysis procedures to ensure the reliable detection of low voltage signals while minimizing voltage response and voltage measurement variability.

   The Braidwood data analysis guidelines were revised prior to A1R04 to be in accordance with "Appendix A" type calibration, recording, and analysis requirements.

4. Implement a data analyst training and qualification program to ensure that voltage measurement variability among different analysts is within the assumed distribution.

   The Braidwood data analyst training program was revised prior to A1R04 to include portions on voltage measurement. A two day training and testing program to ensure the Braidwood analysis guidelines were understood and adhered to was given to all data analysts and monitored by an independent consultant. This same independent consultant monitored activities throughout the job.

5. SG tubes should be removed at each refueling outage to verify both the validity of the empirical correlation and to confirm that the dominant form of tube degradation is ODSCC.

   Braidwood removed four tubes from two different SGs. Refer to Table 2 for the location, bobbin signal and RPC data.

CONCLUSIONS

Based on the extensive work completed by CECo and its contractors, it is concluded that Braidwood Unit 1 can operate for its full Cycle 5 safely and reliably based on the known modes of steam generator tube degradation. The approval of this amendment request will cause no significant negative impact on any system, equipment, or operating mode. CECo will continue its efforts to understand the root cause of the tube degradation and take appropriate effective correct actions to mitigate future degradation. In addition, an extensive program to evaluate tubes removed from the Braidwood Unit 1 SGs will add substantially to the industry data bases and to the assurance of continued safe, reliable operation for Braidwood Unit 1.

G. IMPACT OF THE PROPOSED CHANGE

With the implementation of this proposed license amendment request the Braidwood Unit 1 SGs will still satisfy the requirements of RG 1.121. There will be no significant reduction in the margin of safety to protect the health and safety of the public. 964 SG tubes will remain in service that would have otherwise been removed from service by plugging or repaired by sleeving due to ODSCC at the TSPs. This represents an approximate $2.91M cost savings in SG repairs alone. This will also minimize the RCS loop asymmetries and allow the unit to return to approximately rated thermal power. Additionally, implementation of this proposed license amendment request represents the avoidance of a minimum 18 day critical path outage extension, and the associated replacement power costs. CECo believes this to be the quickest way to return Braidwood Unit 1 to power operation prior to the commencement of CECo's peak load season.

H. SCHEDULE REQUIREMENTS

CECo requests that this proposed license amendment request be approved on an emergency basis. Approval of this proposed license amendment request is required in order to declare the Braidwood Unit 1 SGs operable prior to entering Mode 4, Hot Shutdown. Based on the current outage schedule, Braidwood Unit 1 is predicted to be ready, except for approval of this proposed license amendment request, to enter Mode 4 on Wednesday, April 27, 1994. Approval of this request as soon as possible will allow Braidwood Unit 1 to resume power operations and will help to preclude the possibility that CECo will have inadequate owned reserves entering its peak load period.

FIGURE 1

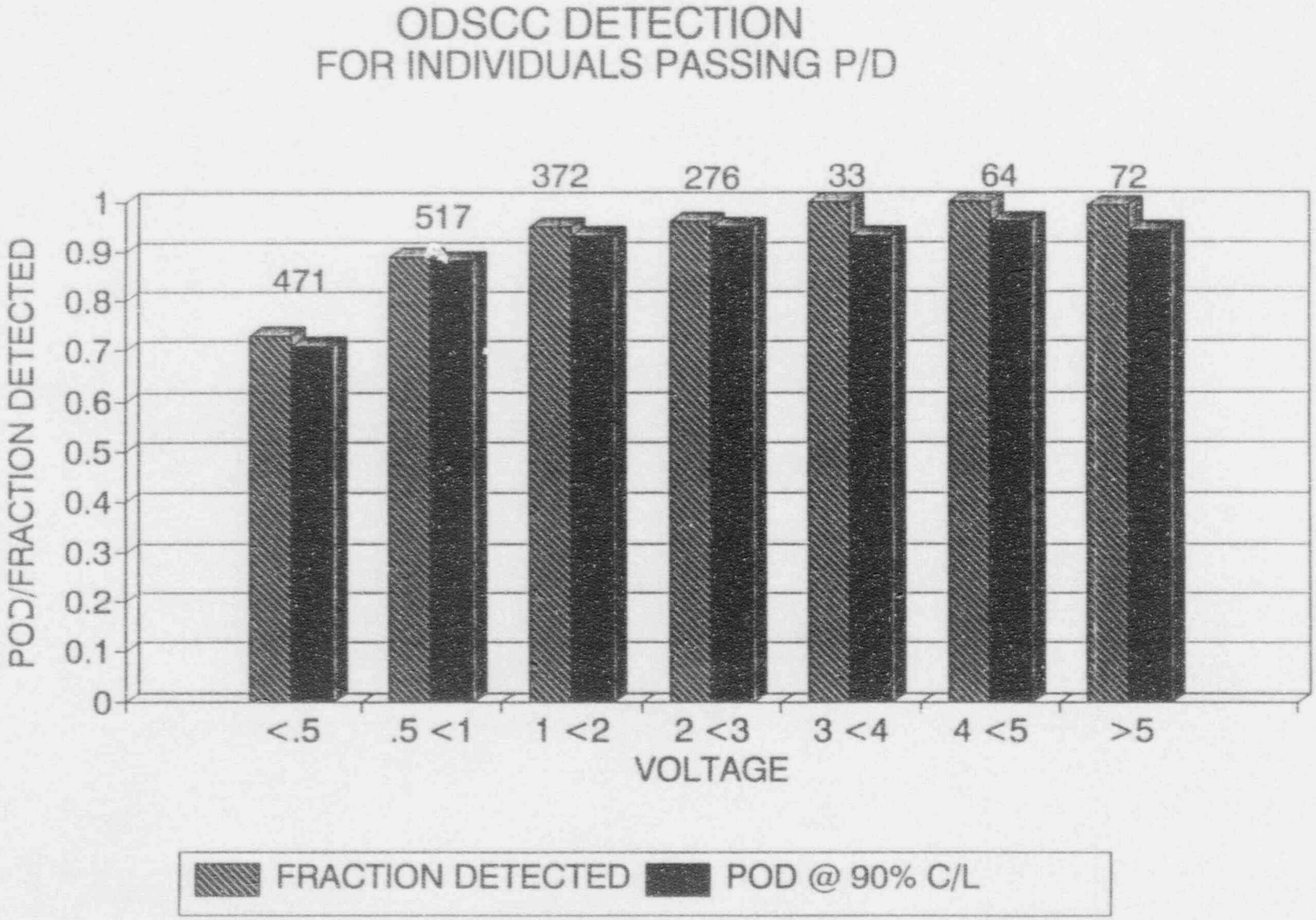
ODSCC DETECTION
FOR INDIVIDUALS PASSING P/D
POD/FRACTION DETECTED
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
471
517
372
276
33
64
72
<.5
.5 <1
1 <2
2 <3
3 <4
4 <5
>5
VOLTAGE
FRACTION DETECTED
POD @ 90% C/L

Table 1: Braidwood 1, SG "D", BOC 5 Probability of Burst Estimate Recommended Database, PoD = 0.6 for V < 3.0 V & 1.0 for V >= 3.0 V						
Volts	INDs	mu.Pb	var.Pb	sd.PbSf	t.Deviates	Pr(Burst)
0.20	4.0	9.516	0.879	1.049	6.628	9.63E-09
0.30	51.7	8.999	0.868	1.033	6.233	6.57E-07
0.40	137.3	8.632	0.862	1.022	5.941	5.89E-06
0.50	180.7	8.348	0.857	1.014	5.708	2.02E-05
0.60	172.3	8.115	0.854	1.008	5.512	4.25E-05
0.70	143.0	7.918	0.852	1.003	5.344	6.90E-05
0.80	114.0	7.748	0.850	0.998	5.197	9.88E-05
0.90	62.3	7.598	0.848	0.995	5.065	9.06E-05
1.00	55.7	7.464	0.847	0.992	4.945	1.28E-04
1.10	33.3	7.342	0.846	0.989	4.836	1.17E-04
1.20	13.0	7.231	0.845	0.986	4.736	6.68E-05
1.30	10.3	7.129	0.844	0.984	4.643	7.54E-05
1.40	6.0	7.035	0.844	0.982	4.556	6.05E-05
1.50	5.0	6.947	0.843	0.980	4.474	6.81E-05
1.60	3.7	6.864	0.843	0.979	4.398	6.61E-05
1.70	4.0	6.787	0.843	0.977	4.326	9.37E-05
1.80	3.0	6.714	0.842	0.976	4.257	9.00E-05
1.90	2.7	6.645	0.842	0.974	4.192	1.01E-04
2.00	2.0	6.580	0.842	0.973	4.130	9.43E-05
2.20	2.7	6.458	0.842	0.971	4.014	1.89E-04
2.30	0.7	6.401	0.841	0.970	3.960	5.70E-05
2.50	0.7	6.295	0.841	0.968	3.857	8.09E-05
2.60	0.7	6.245	0.841	0.967	3.809	9.53E-05
2.80	0.7	6.151	0.841	0.966	3.717	1.30E-04
2.90	0.7	6.106	0.841	0.965	3.674	1.50E-04
3.20	0.0	5.980	0.841	0.963	3.551	0.00E+00
3.30	0.0	5.941	0.841	0.963	3.512	0.00E+00
3.70	0.0	5.795	0.842	0.961	3.368	0.00E+00
3.90	0.0	5.728	0.842	0.960	3.301	0.00E+00
4.00	0.0	5.696	0.842	0.959	3.269	0.00E+00
4.30	0.0	5.604	0.842	0.958	3.177	0.00E+00
5.10	0.0	5.386	0.843	0.955	2.958	0.00E+00
8.90	0.0	4.676	0.848	0.949	2.230	0.00E+00
10.50	0.0	4.465	0.850	0.947	2.011	0.00E+00
					Total:	1.99E-03

Table 2

BRAIDWOOD UNIT 1 TUBE PULLS DURING A1R04

Braidwood removed four tubes from two different S/Gs. the location, bobbin signal and RPC data are as follows:

STEAM GENERATOR "A"

ROW 27	COL 43	INDICATIONS			
		FDB	NDD		
		3-H	88%	4.99 VOLTS	53 DEG
		3-H	SAI	RPC	
		5-H	NDD		
ROW 42	COL 44	INDICATIONS			
		FDB	NDD		
		3-H	68%	3.73 VOLTS	76 DEG
		3-H	MAI	RPC	
		5-H	50%	2.09 VOLTS	93 DEG
		5-H	SAI	RPC	
		7-H	NDD		

STEAM GENERATOR "D"

ROW 16	COL 42	INDICATIONS			
		FDB	NDD		
		3-H	74%	3.12 VOLTS	74 DEG
		3-H	MAI	RPC	
		5-H	DSI*	0.61 VOLTS	51 DEG
		5-H	NDD	RPC	
ROW 37	COL 34	INDICATIONS			
		FDB	NDD		
		3-H	DSI	1.04 VOLTS	65 DEG
		3-H	SAI	RPC	
		5-H	83%	10.44 VOLTS	67 DEG
		5-H	SAI	RPC	

NOTE: ALL INTERSECTIONS WERE INSPECTED WITH RPC PRIOR TO TUBE PULL

* BASED ON RE-EVALUATION IN LAB

TABLE 3

OPERATOR ACTIONS

1 PLANT SHUTDOWN REQUIREMENTS:

a. Leak rates:

Leak Rates	Time to perform plant shutdown
SG leakage greater than 10 gpm	Be shutdown within 4 hours GO TO BWOA SEC-8 step 4. Refer to T.S. 3.4.6.2.
SG leakage greater than 150 gpd but less than 10 gpm	Be shutdown within 10 hours and perform normal cooldown per BWGPs. Refer to T.S. 3.4.6.2.
Leak rate less than 150 gpd	Continue monitoring leak rate.

b. Change in leak rate:

Change in leak rate	Time to perform plant shutdown
Change in any SG leak rate of greater than 100 gpm in 1 hour and leak rate greater than 50 gpd	Be shutdown within 4 hours GO TO BWOA SEC-8 step 4. Refer to T.S. 3.4.6.2.
Change in any SG leak rate of 25 to 100 gpd in 1 hour and leak rate greater than 50 gpd	Be shutdown within 5 hours and perform normal cooldown per BWGPs. Refer to T.S. 3.4.6.2.
Change in any SG leak rate of less than 25 gpd in 1 hour <u>AND</u> leak rate less than 150 gpd	Continue monitoring leak rate.