Docket file

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 18, 1994

Docket No. STN 50-456

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III, Suite 500
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Farrar:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M89697)

The U. S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 54 to Facility Operating License No. NPF-72 for the Braidwood Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your request dated June 20, 1994, as supplemented by your letter dated August 18, 1994.

The amendment revises the TSs by adding a surveillance requirement related to the plant specific inspection guidelines to Section 4.4.5.4.a(11). This surveillance requirement had previously been added as a footnote to this section in Amendment No. 50 to Facility Operating License No. NPF-72, dated May 7, 1994. Additionally, Item 3 of Section 4.4.5.4.a(11) has been revised to remove your previous calculation of primary-to-secondary leakage of 26 gallons per minute (gpm) from the Braidwood, Unit 1, steam generators (SG) under postulated accident conditions at the end of 100 calendar days of operation from restart of Unit 1, Cycle 5. In place of this prior leakage calculation, your revised calculated value of the primary-to-secondary leakage rate of less than 9.1 gpm at the end of Cycle 5 is inserted, including a reference to the basis for this revised estimate. Finally, Section 3.4.8.a is revised to eliminate the time limit on operation in the footnote added in Amendment No. 50 which reduced the TS limit of this concentration to 0.35 microcuries per gram of coolant for the first 100 calendar days of operation. The dose equivalent iodine-131 concentration remains at this value in accordance with the proposal in your letter dated August 18, 1994.

These revisions are consistent with the interim plugging criteria (IPC) that were incorporated into the Unit 1 TSs by Amendment No. 50. In addition, all other revisions that were incorporated into the Braidwood 1 TSs in Amendment No. 50, remain in effect except for the deletion of the limit on operating time as noted above. As noted in our letter of May 7, 1994, the IPC is applicable only for the regions of the SG tubes at the intersections of the tubes with the tube support plates (TSPs). For all other areas of the SG tubes (i.e., in the free span lengths between the TSPs) the TS requirement that SG tubes be removed from service by plugging or be repaired by sleeving if inspection of the SG tubes indicates that the tube wall thickness imperfection depth is equal to 40 percent or more of the nominal wall thickness, remains in effect.

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While this approval of the TS revisions proposed in your letter dated June 20, 1994, does not incorporate any explicit limitation on the operating time of Braidwood, Unit 1, as was done in Amendment No. 50, we require that you conduct a mid-cycle inspection and repair of all four SGs in accordance with the Unit 1 TSs. The purpose of this mid-cycle inspection is to permit a reassessment of the structural and leakage integrity of the SG tubing. The inspection results will provide the basis for this reassessment by determining whether the accelerated growth in eddy current voltage that occurred during the last fuel cycle, is continuing. This mid-cycle inspection should be initiated no later than March 1, 1995. We request that within 30 days of the receipt of this letter, you provide written confirmation that you will perform this mid-cycle inspection. We also request that you submit your plans for assessing the mid-cycle inspection data at least 60 days before your planned shutdown.

On the basis that the staff has confirmed your analysis that the SG leakage is reduced under accident conditions and that the reactor coolant iodine-131 concentration will remain unchanged as proposed in your letter of August 18, 1994, we find that our prior determination of no significant hazards consideration is not affected.

We note that the staff has prepared a draft generic letter providing guidance to licensees choosing to amend their operating licenses to incorporate the IPC into their plant TSs. This draft generic letter has been reviewed internally by the Committee to Review Generic Requirements (CRGR) on July 26, 1994, and by the Advisory Committee on Reactor Safeguards (ACRS) in its meetings on August 3 and August 4, 1994. Both committees recommended release of this draft for public comment, and it was published in the Federal Register on August 12, 1994.

We further note that in connection with the preparation of this draft generic letter, an NRC staff member issued a memorandum containing a Differing Professional Opinion (DPO), dated July 13, 1994. The technical issues raised in this DPO are being evaluated in accordance with NRC policy. The subject DPO has subsequently been released to the Public Document Room (PDR) with permission of its author. Braidwood, Unit 1, was one of several nuclear power plants cited in this DPO. Both the CRGR and the ACRS have considered the subject DPO at their most recent meetings when reviewing the pending draft IPC generic letter.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original Signed By:

Ramin R. Assa, Acting Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 54 to NPF-72
2. Safety Evaluation

cc w/enclosures:
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*See previous concurrence

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NAME	*EBUTCHER	MYOUNG	RCAPRA			
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cc w/enclosures:
See next page

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Unit No. 1

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 54
License No. NPF-72

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated June 20, 1994, as supplemented August 18, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 54 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 18, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 54

FACILITY OPERATING LICENSE NO. NPF-72

DOCKET NO. STN 50-456

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
3/4 4-17	3/4 4-17
3/4 4-17a	3/4 4-17a
3/4 4-27	3/4 4-27

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

10) Tube Repair refers to a process that reestablishes tube serviceability. Acceptable tube repairs will be performed by the following processes:

    a) Laser welded sleeving as described by Westinghouse report WCAP-13698, Rev. 1, or

    b) Kinetic welded sleeving as described by Babcock & Wilcox Topical Report BAW-2045PA, Rev. 1.

    Tube repair includes the removal of plugs that were previously installed as a corrective or preventative measure. A tube inspection per 4.4.5.4.a.8 is required prior to returning previously plugged tubes to service.

11) Tube Support Plate Interim Plugging Criteria Limit for Unit 1 Cycle 5 is used for the disposition of a steam generator tube for continued service that is experiencing ODSCC confined within the thickness of the tube support plates. For application of the tube support plate interim plugging criteria 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 consistent with the eddy current guidelines in Appendix A of WCAP-13854 as appropriate to accommodate the additional information needed to evaluate tube support plate signals with respect to the voltage parameters as specified in Specification 4.4.5.2. Pending incorporation of the voltage verification requirements in 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, provided Item 3 below is satisfied.

    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 and provided Item 3 below is satisfied.

3. The projected end of cycle distribution of crack indications is verified to result in total primary to secondary leakage less than 9.1 gpm (includes operational and accident leakage). The basis for determining expected leak rates from the projected crack distribution is provided in WCAP-14046, "Braidwood Unit 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria" dated May 1994.

4. 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 WCAP-14046, "Braidwood Unit 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria," dated May 1994, shall be excluded from application of the tube support plate interim plugging criteria limit. It has been determined that these tubes may collapse or deform following a postulated LOCA + SSE.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair in the affected area all tubes exceeding the plugging or repair limit) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and**

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

b. With the specific activity of the reactor coolant greater than $100/\bar{E}$ microCuries per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

*With T_{avg} greater than or equal to 500°F.

**For Unit 1 Cycle 5, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.35 microCuries per gram.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 54 TO FACILITY OPERATING LICENSE NO. NPF-72

COMMONWEALTH EDISON COMPANY

BRAIDWOOD STATION, UNIT NO. 1

DOCKET NO. STN 50-456

1.0 INTRODUCTION

By letter dated June 20, 1994, as supplemented on August 18, 1994, the licensee submitted a license amendment request to revise the TSs for Braidwood, Unit 1. The requested amendment would: (1) remove the present 100 calendar day operating limit imposed on Unit 1 in TS Section 3.4.8.a; (2) incorporate some editorial revisions into the TSs in Section 4.4.5.4.a (11); and, (3) revise the SG leakage in Item 3 of TS Section 4.4.5.4.a (11) calculated using the leakage methodology cited in a Westinghouse Report. To support this amendment, the licensee provided information pertaining to the integrity of the SG tubing by letter dated June 10, 1994. The staff's review of this request is provided below.

A proposed determination was published in the Federal Register (59 FR 35839) that the license amendment request involves no significant hazards consideration. This determination was based, in part, both on consideration of the proposed correlation of the conditional SG tube leakage with the bobbin voltage amplitude and a proposed increase of the dose equivalent iodine-131 concentration. The effect of this proposed correlation is to significantly reduce, by a factor of about six, the calculated value of SG leakage under accident conditions while the proposed increase in the maximum permissible iodine-131 concentration would potentially increase the offsite thyroid radiation exposures by about a factor of three. The net effect of these opposing trends would be an overall reduction in the potential offsite radiation exposures.

Subsequently, we have found that a correlation exists between the conditional SG tube leakage and the bobbin voltage amplitude as measured in accordance with the interim plugging criteria (IPC). This finding confirms, in part, our prior determination of a no significant hazards consideration. Furthermore, the proposal in the licensee's letter dated August 18, 1994, to maintain the TS limit on the maximum permissible iodine-131 concentration at 0.35 microcuries per gram of coolant rather than restore it to a higher value, adds an additional degree of conservatism in the potential radiation exposure

doses. On this basis, the proposal to maintain the present value of iodine concentration did not alter our determination of no significant hazards consideration.

2.0 BACKGROUND

By letters dated April 25, April 28, April 30, May 2, May 4, and May 6, 1994, Commonwealth Edison Company (ComEd or the licensee), submitted a request to change the technical specifications for Braidwood Station, Unit 1. The requested amendment revised, in part, Technical Specifications (TSs) 3.4.6.2, 3.4.8, 4.4.5.2, 4.4.5.4 and 4.4.5.5 and Bases Section 3/4.4.5 to allow the usage of a voltage-based steam generator tube plugging criteria for defects located at the tube support plate (TSP) elevations. Some of the proposed changes were applicable only for the fifth operating cycle; the reduction in the dose equivalent iodine-131 concentration in the reactor coolant was limited to the first 100 days of operation in the present fuel cycle.

The staff reviewed the submittals cited above and approved the proposed TS revisions in its letter dated May 7, 1994, "Issuance of Amendment No. 50 to Facility Operating License NPF-72, Braidwood Station, Unit 1 (TAC No. M89166)." The staff concluded in its Safety Evaluation (SE) issued in conjunction with Amendment 50 that the proposed interim tube repair limits and the reduced operating leakage limits would ensure adequate structural and leakage integrity of the steam generator (SG) tubing for the first 100 calendar days from the date of restart of Unit No. 1. Operation of the plant with a primary coolant hot leg temperature below 500 °F did not need to be counted as part of the 100 calendar days.

The voltage-based SG tube repair criteria approved for Braidwood 1 in Amendment No. 50, applies specifically to flaws identified as outside diameter stress corrosion cracking (ODSCC) and which are confined within the thickness of the TSPs; this approach provides an alternative to the existing depth-based TS limit for SG tube flaws. This latter limit requires that SG tubes be removed from service by plugging or be repaired by sleeving in the affected area if inspection of the SG tubes indicates that the tube wall imperfection depth is equal to 40 percent or more of the nominal wall thickness. In part, the voltage-based criteria: (1) permits flaws within the bounds of the TSP elevations with bobbin voltages less than or equal to 1.0 volt to remain in service; (2) permits flaws within the bounds of the TSP with bobbin voltages greater than 1.0 volt but less than or equal to 2.7 volts to remain in service if a rotating pancake coil (RPC) probe does not detect degradation; and (3) requires flaw indications at the TSP elevations with bobbin voltages greater than 2.7 volts to be plugged or repaired.

The licensee's current proposal submitted in its letter dated June 20, 1994, would modify, in part, the Braidwood, Unit 1, TSs to remove the 100 calendar day limit on the operating interval of Unit 1. The 100 day limit was imposed as a consequence of the calculated primary-to-secondary leakage which could occur in the event of a main steamline break (MSLB) using the method recommended by the staff in draft NUREG-1477, "Voltage-Based Interim Plugging

Criteria for Steam Generator Tubes," dated June 1993, and the dose equivalent iodine-131 limit of 0.35 microcuries per gram of coolant as proposed by the licensee in its prior license amendment request on this matter. In its letter dated June 20, 1994, the licensee proposed that this iodine-131 concentration be returned to its prior value of 1.0 microcuries per gram of coolant but subsequently requested in its letter dated August 18, 1994, that this concentration remain at 0.35 microcuries per gram. The licensee's current request for a license amendment provides an alternative to the draft NUREG-1477 calculation methodology for SG tube leakage resulting from a postulated MSLB which involves demonstrating that a linear correlation exists between the logarithms of the SG tube leak rate and the associated bobbin voltage.

The staff's detailed evaluation of the present amendment request is contained in Section 3.0 in which the inspection issues associated with the voltage-based criteria are evaluated in Section 3.1. An overview of tube integrity issues related to SG tube leakage and burst are discussed in Section 3.2. Section 3.3, in turn, deals specifically with the SG tube structural integrity issue (i.e., burst) under postulated accident conditions while Section 3.4 addresses the issue of SG tube leakage under normal operating and postulated accident conditions. The radiological consequences of SG tube leakage under accident conditions is evaluated in Section 3.5. The evaluation of the SG tube structural integrity from a risk based perspective is presented in Section 3.6. The leakage monitoring and administrative control enhancements implemented by the licensee is evaluated in Section 3.7 and the issue of limited TSP deflections under accident conditions as proposed by the licensee, is contained in Section 3.8. Finally, Section 4.1 of this SE provides a summary of the technical evaluations presented in Section 3.0 while Section 4.2 provides the basis for our acceptance of the proposed TS revisions.

3.0 EVALUATION

3.1 Inspection Issues

The staff's evaluation of the licensee's inspections with respect to the voltage-based repair criteria was documented in the SE issued with Amendment No. 50. In summary, the plant specific guidelines used at Braidwood, Unit 1, during the Spring 1994 inspections were consistent, as appropriate, with those contained in Appendix A of the Westinghouse Report, WCAP-13854, "Technical Support for Cycle 8: Steam Generator Tube Interim Plugging Criteria for Catawba, Unit 1," dated September 1993. Appendix A of WCAP-13854 was referenced in Amendment Nos. 111 and 105 for Catawba, Units 1 and 2, respectively, which were issued on December 16, 1993. The inspection guidelines contain, in part, the bobbin specifications, calibration requirements, specific data acquisition and analyses criteria, and flaw recording guidelines which were used in the inspection of the SG tubing at Braidwood, Unit 1, during the Spring 1994 refueling outage.

3.2 Tube Integrity Issues

The purpose of the TS tube repair limits is to ensure that SG tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions, consistent with General Design Criteria (GDC) 14, 15, 31, and 32 of Appendix A to 10 CFR Part 50. Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the SG tubing. Leakage integrity refers to limiting the primary-to-secondary SG leakage to within limits which will not result in offsite radiation doses exceeding a small fraction of the guideline values in 10 CFR Part 100. The traditional approach for accomplishing these objectives has been to establish a minimum wall thickness requirement in accordance with the structural criteria of Regulatory Guide (RG) 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes." Allowance for eddy current measurement error and flaw growth between inspections has been added to the minimum wall thickness requirements, consistent with RG 1.121, to arrive at a depth-based repair limit. Enforcement of a minimum wall thickness requirement implicitly serves to ensure leakage integrity during normal operation and accidents, as well as structural integrity. It has been recognized, however, that defects, especially cracks, may occasionally grow entirely through-wall and develop small leaks. For this reason, limits on the allowable primary-to-secondary leakage have been established in a plant's TSs to ensure timely plant shutdown before adequate structural and leakage integrity of an affected SG tube is impaired.

The interim SG tube repair limits identified as the interim plugging criteria (IPC) for Braidwood, Unit 1, consist of voltage amplitude criteria at TSP intersections rather than the traditional depth-based criteria. Thus, the repair criteria at these intersections represents a departure from the past practice of explicitly enforcing a minimum wall thickness requirement.

The industry-wide data base from pulled tube examinations have shown that for bobbin voltage indications exceeding 1.0 volt (i.e., the IPC lower repair limit for Braidwood 1), crack depths have ranged between 50% and 100% through-wall. The likelihood of through-wall or near through-wall crack penetrations appears to increase with increasing voltage amplitude. For bobbin voltage indications at or near 2.7 volts, the maximum crack depths have been found to generally range between 90% and 100% through-wall. Many of the SG tubes which will be allowed to remain in service under the IPC may have, or may develop, through-wall or near through-wall crack penetrations during the present fuel cycle (i.e., Cycle 5), thus creating the potential for leakage during normal operation and a postulated MSLB accident. The staff's evaluation of the repair criteria from a structural and leakage integrity standpoint is provided in Sections 3.3 and 3.4 of this SE.

3.3 Structural Integrity

3.3.1 Deterministic Structural Integrity Assessment

The licensee provided in Attachment F of its submittal dated April 25, 1994, and subsequently provided in the Westinghouse Report, WCAP-14046, "Braidwood, Unit 1: Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria," dated May 1994 (Proprietary) submitted with its letter dated June 10, 1994, a burst pressure versus bobbin voltage correlation demonstrating that bobbin indications satisfying the 1.0 volt IPC would retain adequate structural margins, consistent with the structural criteria of RG 1.121. The correlation is developed from both pulled tube data from other plants (using pre-pull bobbin voltages) and laboratory tube specimens containing ODSCC flaws. The calibration standard voltage set-ups and voltage measurement procedures at Braidwood, Unit 1, were conducted in a manner consistent with the bobbin voltage data acquisition methodology used to construct the burst pressure/bobbin voltage correlation.

During the Spring 1994 outage, the licensee pulled portions of four SG tubes containing 13 TSP intersections from the Unit 1 SGs for leak and burst testing and for destructive metallographic examination. These tests and examinations are being performed to: (1) confirm the expected axial ODSCC morphology; (2) enhance the supporting data bases for the burst and leakage correlations; and (3) provide insight into the relative sensitivities of the bobbin and RPC probes for less developed areas of ODSCC. The licensee stated in its letter dated June 20, 1994, that the data from the leak and burst tests on the pulled SG tubes will be provided to the NRC by August 12, 1994.

Since the results of the leak and burst tests conducted by the licensee on the pulled tube specimens removed during the last refueling outage were scheduled to be submitted just prior to completion of the staff's review of the licensee's pending license amendment, the staff held a conference call with the licensee on August 10, 1994, to discuss on a preliminary basis, the results of the pulled tube examinations. The purpose of this discussion was to determine if any of the data from the leak and burst tests departed significantly from prior test data. During this conference call, the licensee stated that the measured burst pressures and leak rates from its pulled SG tube specimens were, in general, consistent with what was expected based on the existing correlations. The licensee further stated that the specimens which were burst tested all failed axially. The licensee also stated that the metallographic examination of the pulled SG tubes which will confirm the nature of the degradation at the support plate elevations, had not yet been completed. As a result, the licensee revised its submittal date for its pulled SG tube examination report to the end of August 1994. However, the preliminary results indicate that, in general, the degradation at Braidwood, Unit 1, behaves similarly to that observed at other plants for which the voltage-based limits have been approved. The staff requested during this conference call that they be notified if any unexpected results are found during the remaining destructive analyses.

The interim voltage repair criteria approved by the staff for several plants have been set deterministically to ensure that SG tubes will retain adequate structural integrity during the full range of normal, transient, and postulated accident conditions with allowances for eddy current test uncertainty and flaw growth during the next operating cycle. Because ODSCC is confined within the thickness of the TSPs during normal operation, the staff has concluded that the structural constraint provided by the TSPs ensures that all SG tubes with ODSCC, irrespective of the voltage amplitude of the indication, will retain a margin of safety of three with respect to burst under normal operating conditions, consistent with the criteria of RG 1.121. For a postulated MSLB accident, however, the TSPs may displace axially during blowdown such that the ODSCC affected portion of the SG tubing may no longer be fully constrained by the TSPs. Thus, the staff has conservatively considered the ODSCC affected regions of the SG tubes as free standing tubes for the purpose of assessing burst integrity under postulated MSLB conditions.

The allowable end-of-cycle (EOC) voltage which ensures a margin of 1.43 with respect to burst under postulated MSLB conditions in accordance with RG 1.121, is based on the lower 95% prediction interval of the burst pressure/bobbin voltage correlation, adjusted for lower bound material properties evaluated at the 95/95 confidence level. This voltage is about 4.5 volts for the 3/4-inch diameter tubing used in the SGs of Braidwood, Unit 1. The difference between the 4.5 volt allowable EOC voltage and the 1.0 volt repair criterion represents an allowance of 3.5 volts for voltage growth during the present fuel cycle and for eddy current voltage measurement variability (i.e., the repeatability error) during the last SG inspection in Spring 1994.

For Braidwood, Unit 1, the maximum EOC voltages for Cycle 5 are expected to be dominated by the contribution of voltage growth during the course of the present fuel cycle, based on the voltage growth rates observed during the previous cycle. The last fuel cycle for Unit 1 (i.e., Cycle 4) lasted for 1.147 effective full power years (EFPY) while the present fuel cycle (i.e., Cycle 5) will be about 1.231 EFPY. The observed growth rates in Cycle 4 ranged up to a maximum of 9.8 volts per cycle. The licensee expects that improved molar ratio chemistry control (i.e., the ratio of sodium to chloride) and boric acid addition will reduce the present and future corrosion rates below those of the last fuel cycle. The staff notes, however, that any such improvement is difficult to quantify. For this reason, it is appropriate to assume that growth rates in the present fuel cycle will be similarly distributed as those observed in the previous fuel cycle. Due to the growth rates observed at Braidwood, Unit 1, in the last fuel cycle, the contribution of the initial beginning-of-cycle (BOC) bobbin voltage amplitude to the maximum EOC bobbin voltage is relatively small for a 1.0 volt repair criterion. The contribution of voltage measurement variability to the EOC voltage is also relatively small, with the upper 95% confidence value estimate, based on industry data, equal to 20% of the initial voltage measurement (i.e., $\leq$ 0.2 volts). Thus, voltages could range to a maximum of between 9.8 and 11 volts by the end of the present operating cycle, well in excess of the allowable EOC voltage limit of 4.5 volts cited above for 3/4-

inch diameter SG tubing. The allowable EOC voltage limit could be satisfied by limiting operation of Unit 1 during the present fuel cycle to 4.6 months after restart in May 1994.

The 4.6 month operating interval cited above was determined by evaluating the time it takes a flaw indication to grow to the point that the structural voltage limit of 4.5 volts is exceeded. In this analysis, a 1.0 volt bobbin indication is assumed to grow at a rate equal to the maximum growth rate observed in the prior cycle (i.e., 9.8 volts per 1.147 EFPY) and that the indication has been undersized by 20% (i.e., the 95 percent cumulative probability on the non-destructive examination (NDE) uncertainty). The determination of the structural voltage limit is determined from a correlation of burst pressure as a function of bobbin voltage. The deterministic burst pressure is evaluated at the lower 95 percent prediction interval which has been adjusted for SG tubing with lower 95/95 material properties at a pressure of 1.43 times the differential pressure expected during an MSLB. The maximum differential pressure under accident conditions is calculated as 2560 pounds per square inch (psi); the SG tube burst pressure voltage is, therefore, evaluated at a differential pressure of 3660 psi. This evaluation results in a structural voltage limit of about 4.5 volts. For the conditions discussed above, a 1.0 volt indication would still have margin against burst under MSLB differential pressure conditions beyond the 4.6 months though it would be lower than 1.43. For example, operation of Unit 1 for about 10 months after restart in May 1994 would result in a margin of about 1.15 against burst under MSLB differential pressure conditions. The deterministic analyses described above have limitations as discussed in Section 3.3.2. As a result of these limitations, a probabilistic structural integrity assessment has been performed and the staff's evaluation is presented in Section 3.3.2. An assessment of other aspects relevant to operation of Braidwood, Unit 1, with SG tube ODSCC is presented in Sections 3.5 and 3.7.

3.3.2 Probabilistic Structural Integrity Assessment

The licensee has performed a probabilistic analysis of the potential for SG tube rupture, given an MSLB. The need for this analysis, which supplements the deterministic analysis discussed above, is dictated by the following considerations:

1. The deterministic analysis does not consider the tail of the burst pressure distribution beyond the lower 95% prediction interval used to determine the maximum allowable EOC voltage. Given the large numbers of indications being accepted for continued service with the 1.0 volt criterion, the probabilistic analysis ensures that the use of the 95% prediction interval value in lieu of the 99% or 99.9% values does not lead to a significant likelihood of SG tube rupture given an MSLB.

2. The deterministic assessment ignores the burst and leakage potential of bobbin indications between 1.0 volt and 2.7 volts for which the RPC probe failed to confirm the indication. The probabilistic assessment, however, considers the burst potential of these indications with no

credit given for the lack of confirmation by the RPC probe of the presence of these indications.

3. The deterministic analysis does not account for bobbin indications missed by the data analysts. The staff concluded in draft NUREG-1477 that the probabilistic assessment is required in order to address the burst potential of indications missed by the IPC data analysts.

4. The deterministic analysis does not consider the cumulative effect of the entire distribution of indications accepted for continued service. Employing the probabilistic analysis, however, ensures that all indications accepted for continued service are accounted for in determining the overall probability of burst given an MSLB.

5. The deterministic analysis does not consider the tails of the material properties distribution and the eddy current voltage variability distributions. The probabilistic analysis does include the entire distribution of material properties and voltage variability.

To perform the probabilistic analysis, the EOC distribution of indications must be determined. Consistent with the approach recommended in draft NUREG-1477, the BOC distribution used in the determination of the EOC distribution involves adjusting the indications detected during the inspection by the probability of detection (POD), where the POD is assumed to have a constant value of 0.6, irrespective of voltage. The net effect of this assumption is that the distribution of detected bobbin indications is scaled up by a factor of 1/POD. After the POD scaling is made, indications removed from service by tube repair (i.e., plugging or sleeving) are subtracted from this distribution to yield the assumed BOC distribution. The EOC distribution is then determined by combining the voltage measurement uncertainty distribution, the voltage growth rate distribution, and the BOC voltage distribution using Monte Carlo techniques. For each of the resultant EOC voltages determined by the above analysis, the distribution of burst pressures as a function of bobbin voltage along with a distribution of material properties is sampled by Monte Carlo techniques to yield a frequency density function of burst pressures for that voltage.

The POD scaling approach cited above is reasonably consistent with reported operating experience to date with ODSCC in terms of accounting for the projected distribution of indications at EOC which were not previously detectable at BOC. However, with the exception of Braidwood, operating experience to date with SG tube ODSCC at TSP intersections is that maximum EOC bobbin voltages generally do not exceed 4 or 5 volts. Although there are known cases where indications on the order of 3 volts have not been detected, there is very little experience regarding the likelihood of not detecting indications between 3 and 10 volts. The assumption of a 0.6 POD for indications between 3 and 10 volts for Braidwood 1 leads to a calculated conditional probability of rupture for a postulated MSLB of 3.1E-2 at BOC and 9E-2 at EOC. The licensee believes that the numerical value of the POD is substantially higher than 0.6 for indications exceeding 3 volts, based on data

collected from the Electric Power Research Institute (EPRI) performance demonstration program and the licensee's reanalysis of previous eddy current inspection data.

Preliminary staff review of the results from EPRI's performance demonstration program identified several concerns; most notable of these is the absence of an objective benchmark for evaluating the performance of an individual analyst. As a result, pending further staff review of the results from the EPRI performance demonstration program, the staff believes that a POD of 0.6 is appropriate for Braidwood, Unit 1.

The licensee anticipates obtaining additional insights on the detectability of large amplitude indications based on further study of the results of the recently completed SG inspections at Braidwood, Unit 1. The assumption of a POD equal to 1.0 for indications exceeding 3 volts leads to a conditional probability of rupture estimate for a postulated MSLB of 2E-3 at BOC and 5E-3 at EOC, which are less than the conditional probability estimates considered in the analysis of risk contained in NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." The use of a POD value higher than 0.6 is discussed below.

In the absence of significant noise, it is difficult to understand why a field analysis, with its attendant quality assurance procedures, would fail to detect an ODSCC defect in the range of 3 to 10 volts. However, the same logic is true for voltage indications in the 1 to 3 volt range and experience has shown that there is a potential for missing indications in this range. The staff concludes that this is an area requiring further study. Meanwhile, the limited evidence provided by the licensee does not provide an adequate basis for assuming a POD equal to 1 for indications exceeding 3 volts. The risk implications of the conditional probability of burst estimates associated with the 0.6 POD assumption are assessed in Section 3.6 of this evaluation.

The licensee has also performed a probability of burst analysis that considers the constraining influence of the TSPs in mitigating the potential for burst during a postulated MSLB. The licensee believes that the evaluation of the limited TSP displacements provides a more realistic approach than that described above. The licensee's analysis includes an assessment of the displacement of the support plate during an MSLB and the portion of an ODSCC defect which may be exposed (i.e., unconstrained) as a result of this displacement. Accordingly, the licensee's assessment evaluates the potential for rupture of a given crack as a function of its exposed length outside the thickness of the TSPs when the support plate is assumed to be in its displaced configuration. This analysis leads to an extremely low estimate of the probability of burst of 3.1E-5 at EOC in the event of an MSLB. The NRC staff has not completed its review of the TSP displacement analysis and has not reviewed the methodology for calculating the probability of burst given that the TSP displacement is limited. Therefore, the staff's evaluation does not rely on the results of the licensee's analysis taking credit for limited displacement of the TSPs. A discussion of the staff's continuing review of

the analyses of the TSP deflections under accident conditions is presented in Section 3.8.

3.3.3 Data Exclusion from the Burst Pressure Correlation

During the performance of the pulled tube examinations, malfunctions in the test equipment or improper specimen preparation can occasionally occur which could result in erroneous readings. Data such as this should not be included in the data base since it could result in invalid results and/or conclusions. The staff, therefore, concluded in draft NUREG-1477 that eliminating data from the IPC burst pressure data base was appropriate provided that the data could be shown to be erroneous or the result of an invalid test. The staff provided additional guidance regarding the exclusion of data from the correlations used in the IPC burst pressure data base in a meeting with the industry on February 8, 1994. As a result of this guidance, the licensee provided criteria for determining whether data may be removed from the burst pressure versus bobbin voltage correlation. The specific criteria are presented in Tables 5-1, 5-2, and 5-3 of the Westinghouse Report, WCAP-14046, which was attached as an enclosure to the licensee's letter dated June 10, 1994.

The data points excluded from the burst pressure correlation as a result of applying these criteria are listed in Table 5-4 of the subject document. The staff has concluded that excluding the data points listed in Table 5-4 from the 3/4-inch diameter SG tubing burst pressure data base is appropriate. Pending further evaluation of the generic criteria presented in Tables 5-1, 5-2, and 5-3, the staff is continuing to assess the appropriateness of excluding data points from the burst pressure correlation on a case-by-case basis.

3.4 Leakage Integrity

3.4.1 Normal Operational Leakage

An important implication of voltage-based SG tube repair criteria is that the IPC criteria may permit SG tubes with up to 100% through-wall cracks to remain in service. Thus, the leakage integrity of these SG tubes, in addition to their structural integrity, must be assessed. Adequate leakage integrity during normal operation is implicitly ensured by the allowable limits on the operational SG leak rate in the plant TSs. As part of the IPC approved in Amendment No. 50, the licensee implemented more restrictive operational primary-to-secondary SG leakage limits than those previously authorized. Specifically, the Unit 1 TSs, as revised in Amendment No. 50, restrict the SG primary-to-secondary leakage through any one SG to 150 gallons per day (gpd) and the total primary-to-secondary leakage through all SGs to 600 gpd. Prior to this revision, the Unit 1 TS limits were 500 gpd through any one SG and 1.0 gallon per minute (gpm) or 1440 gpd through all four SGs.

The SG TS leakage limits implemented in Amendment No. 50 as part of the IPC, provide added assurance against tube rupture since the tube material properties are consistent with a leak-before-break approach. These limits are

also intended to provide additional assurance of timely plant shutdown in the event of any unexpected growth of ODSCC outside the TSPs or unexpectedly high crack growth rates leading to excessive SG leakage under normal operating conditions. On this basis, the staff reaffirms its prior findings in the SE issued in conjunction of Amendment No. 50 that the normal operating primary-to-secondary leak rate limits are acceptable.

3.4.2 Accident Leakage

In Amendment No. 50, operation of Braidwood, Unit 1, was limited to 100 calendar days when the hot leg temperature was above 500 °F as a result of determining that the leakage during a postulated MSLB at the end of the operating cycle was 47.2 gpm using the leak rate methodology described in draft NUREG-1477 (i.e., a constant leak rate model). The licensee has subsequently proposed a model for calculating the SG tube leakage from the faulted steam generator during a postulated MSLB in the enclosure to its letter dated June 10, 1994. The model consists of two major components: (1) a model predicting the probability that a given indication will leak as a function of voltage (i.e., the probability of leakage (POL) model); and (2) a model predicting leak rate as a function of voltage, given that leakage occurs (i.e., the conditional leak rate model).

Regarding the POL model, the licensee in its license amendment request on April 25, 1994, fitted six different functional forms to the POL data, consistent with the approach recommended in draft NUREG-1477. The staff had previously required that all six functional forms for the POL model be considered in the leak rate calculations performed in support of the voltage-based repair criteria approved at several plants. This position reflected the fact that the different functional forms each exhibited acceptable goodness-of-fit with the available data. However, the use of any one of the six statistical models could vary the calculated total leak rate by factors of 2 to 6 with the actual factor depending on the distribution of EOC flaw indications. The staff reaffirms its prior conclusion stated in Amendment No. 50 that use of a single functional form, the log-logistic, is acceptable for the purpose of assessing MSLB-induced SG tube leakage. The staff continues to believe that any non-conservatism associated with the use of the log-logistic model, as compared to the other functional forms, is small compared to the conservatism inherent in the existing methodology for calculating the SG tube leakage and the radiological consequences of this leakage induced by a postulated MSLB. In fact, using the hybrid linear leak rate methodology described below, the licensee observed that the primary-to-secondary leakage calculated for Braidwood, Unit 1, during postulated accident conditions is essentially independent of the POL correlation.

Regarding the conditional leak rate model, the licensee has proposed a correlation between the SG tube leak rate and bobbin voltage based on a linear regression fit of the logarithms of the corresponding SG tube leak rate and voltage data. The staff previously concluded in draft NUREG-1477 that no proven relationship between SG tube leak rate and bobbin voltage existed.

Although this conclusion applied to both 3/4-inch and 7/8-inch diameter SG tubing, it was primarily based on a quantitative analysis of the leak rate data for 7/8-inch diameter SG tubing. However, based on public comments received on draft NUREG-1477, the staff developed a criteria for determining by standard statistical tests whether a correlation between the SG leak rate and bobbin voltage exists. These criteria were presented by the NRC at a meeting at NRC Headquarters in Rockville, Maryland, on February 8, 1994, and would permit the use of a linear relationship between the logarithms of the leak rate and bobbin voltage if the correlation could be statistically justified at a 95% confidence level (i.e., a p-value of 5%). The licensee has proposed that for 3/4-inch diameter SG tubing, such a correlation between SG tube leak rate and bobbin voltage exists based on the acceptance criteria presented during the February 8, 1994, meeting. The staff has reviewed the licensee's analysis and confirmed that such a correlation between the logarithms of the leak rate and bobbin voltage exists for the data base presented in Table 5-10 of WCAP-14046. As a result, the staff concludes that a linear relationship between the logarithms of the leak rate and bobbin voltage can be used in the determination of the primary-to-secondary SG tube leakage during a postulated MSLB.

The licensee has proposed using several different methods for determining the primary-to-secondary SG leakage during a postulated MSLB: (1) a hybrid model that combines Monte Carlo techniques and deterministic calculations; (2) a full Monte Carlo method which simulates the regression parameter uncertainties; and (3) a methodology proposed and described by EPRI in its report, TR-100407, Revision 1 (draft), "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at TSPs," dated August 1993, which determines an EOC leak rate table as a function of BOC voltage amplitude and also accounts for parametric uncertainties. Based on staff analysis of the three models, the staff has concluded that the most appropriate model for the evaluation of the Braidwood 1 SGs is the full Monte Carlo method which simulates the parametric uncertainties (i.e., Item 2 cited above). The staff will continue to evaluate the appropriateness of the other two models as part of its generic activities with respect to SG degradation specific management.

The full Monte Carlo method for determination of the primary-to-secondary leakage during a postulated MSLB involves:

1. Determining random versions of the POL and leak rate correlations to account for the uncertainty in the regression parameters (i.e., parameter uncertainty).

2. Using the regression parameters from Step 1 to determine the leak rate for each flaw indication in the estimated EOC voltage distribution. The EOC voltage distribution used in this calculation is the same as that discussed in Section 3.3.2.

3. Calculating the sum of the SG tube leak rates determined in Step 2 to obtain a value of the total SG tube leak rate.

4. Repeating Steps 1, 2, and 3 many times (e.g., 10,000) to obtain a distribution of the total SG leak rates.

5. Ordering the distribution of total leak rates in Step 4 in ascending order, and taking the 95th quantile at a 95% confidence level as the primary-to-secondary SG leakage during a postulated MSLB. This is the value used in assessing the SG leakage integrity of the SG tubing.

The full Monte Carlo methodology described above was performed by the licensee for the limiting SG which was determined by the licensee to be the "D" SG. The licensee's calculated leakage rate was 3.2 gpm during a postulated MSLB at EOC at a 95% confidence bound. The staff performed independent calculations using the licensee's leak rate data base which confirmed the results of the licensee's calculation. However, as discussed in Section 3.4.3, the staff has also reviewed the leak rate data base and determined that it should be adjusted, thereby resulting in the staff calculating a higher SG leak rate than that calculated by the licensee. The staff notes that the leakage calculated by the licensee using the hybrid methodology and the EPRI methodology cited above as Items 1 and 3, respectively, was 3.1 gpm.

The licensee stated in its submittal dated April 25, 1994, that the allowable SG leak rate was 9.1 gpm in the faulted SG, which was consistent with maintaining the radiological consequences of a release outside containment to within a small fraction of the guideline values in 10 CFR Part 100 based on an assumed initial coolant Iodine activity of 1.0 microcurie per gram of coolant, an accident generated Iodine spike consistent with the appropriate Standard Review Plan, and the use of thyroid dose conversion factors consistent with those promulgated by the International Commission on Radiation Protection in its standard, ICRP-30.

The staff's calculated SG leakage rate at EOC is presented in Section 3.4.3 and the staff's evaluation of the radiological effects is presented in Section 3.5

3.4.3 Staff Evaluation of the End-of-Cycle Steam Generator Leak Rate

The staff agrees that the calculated SG leak rate at EOC is significantly lower than the previously calculated value of 47.2 gpm, but does not agree that it is as low as that calculated by the licensee, as discussed in this section.

During the performance of the pulled tube examinations, malfunctions in the test equipment or improper specimen preparation can occasionally occur which could result in erroneous readings. Data such as this should not be included in the data base since it could result in invalid results and/or conclusions. The staff, therefore, concluded in draft NUREG-1477 that eliminating data from the SG tube leak rate data base was appropriate provided that the data could be shown to be erroneous or the result of an invalid test. The staff provided additional guidance regarding the exclusion of data from the correlations used in the SG tube leak rate data base in a meeting with the industry on

February 8, 1994. As a result of this guidance, the licensee provided criteria for determining whether data may be removed from the SG tube leak rate versus bobbin voltage correlation. The specific criteria are presented in Tables 5-1, 5-2, and 5-3 of the Westinghouse Report, WCAP-14046, which was attached as an enclosure to the licensee's letter dated June 10, 1994.

The data points excluded from the conditional leak rate correlation and the probability of leakage correlation as a result of applying these criteria are listed in Tables 5-5 and 5-6 of WCAP-14046, respectively. The staff has concluded that excluding the data points listed in Table 5-5, with the exception of model boiler specimen 598-3, from the 3/4-inch conditional leak rate data base is appropriate. Furthermore, the staff has concluded that excluding the data points listed in Table 5-6 from the 3/4-inch diameter SG tubing POL data base is appropriate. Pending further evaluation of the generic criteria presented in Tables 5-1, 5-2, and 5-3, the staff is continuing to assess the appropriateness of excluding data points from the conditional leak rate correlation and the probability of leakage correlation on a case-by-case basis.

The licensee provided an analysis of the SG tube leakage expected from an indication on tube R28C41 removed from another nuclear power plant (identified as Plant S). The leakage from this specimen exceeded the capacity of the test facility and, as a result, an assessment of the leakage from this specimen was performed. Pending further staff evaluation of the appropriate leakage value for this data point, the staff has concluded that this data point should be assigned a leakage value of 2496 liters per hour (l/hr) consistent with the leakage predicted using the CRACKFLO model.

The staff has assessed the inclusion of both model boiler specimen 598-3 and pulled tube specimen R28C41 with a conditional leak rate of 2496 l/hr in the conditional leak rate data base. The staff concludes that with this amended data base, a linear correlation between the logarithms of the leak rate and bobbin voltage exists. On this basis, the staff calculates that the primary-to-secondary SG tube leakage at a 95% confidence bound, is 6.8 gpm at EOC. While the staff's calculated SG leak rate is higher than the value of 3.2 gpm calculated by the licensee, this higher value is acceptable in that it would not cause offsite radiation exposures which exceed a small fraction of the guideline exposures of 10 CFR Part 100 as discussed below. As a result, the staff has concluded that operation of Braidwood, Unit 1, need not be limited to 100 calendar days as discussed in Section 3.5 in which the potential radiological offsite consequences of a postulated MSLB at EOC is found acceptable. Accordingly, from a leakage integrity standpoint only, operation of Braidwood, Unit 1, for the remainder of the present fuel cycle is acceptable. However, as noted in Section 3.6, structural integrity concerns control the determination of the appropriate operating interval for Braidwood, Unit 1.

3.5 Radiological Consequences

In Section 3.4.3, the staff concluded that the licensee's calculated value of the primary-to-secondary leakage of 3.2 gpm, using the full Monte Carlo methodology, during a postulated MSLB at EOC for Unit 1 is too low based on the exclusion by the licensee of certain data from the data base. Accordingly, the staff has independently evaluated the radiological consequences of a postulated MSLB using its calculated primary-to-secondary leak rate of 6.8 gpm and the reactor coolant dose equivalent iodine-131 concentration of 0.35 microcuries per gram of coolant which was proposed by the licensee in its letters dated April 25 and August 18, 1994, and is the present TS limit for Cycle 5. The staff used the licensing basis value for X/Q of 5.6×10^{-4} sec/m^3 in performing its evaluation of the radiological consequences of this event at the exclusion area boundary (EAB). The staff used the dose conversion factors for iodine isotopes set forth in the International Commission on Radiation Protection Standard, ICRP 30, as well as the breathing rates set forth in RG 1.4. The radiation exposure doses were also evaluated at the maximum SG leakage projected by the licensee in its proposed revision to the TSs (i.e., less than 9.1 gpm).

Table 1 presents the thyroid doses calculated by the staff for both the pre-existing spike case and the event-generated spike case. The calculated radiation doses to the thyroid at the EAB are listed in Table 1 for both the staff's estimated leakage of 6.8 gpm and the licensee's TS value of less than 9.1 gpm. The whole-body dose acceptance criteria listed in this table are for the pre-accident (i.e., 25 rem) and event-generated (i.e., 2.5 rem) iodine spike cases.

TABLE 1

CALCULATION TYPE AND ACCEPTANCE CRITERIA	THYROID DOSE (REM) PRE-ACCIDENT IODINE SPIKE CASE	THYROID DOSE (REM) EVENT-GENERATED IODINE SPIKE CASE	WHOLE-BODY DOSE (REM)
EAB (2 hr) [6.8 gpm]	18	3	≤0.1
EAB (2 hr) [9.1 gpm]	24	4	≤0.1
ACCEPTANCE CRITERIA	300	30	25/2.5

As can be seen from this table, all the calculated thyroid doses are well within the exposure guideline values of 10 CFR Part 100 for the pre-existing iodine spike case and for the event-generated Iodine spike case, thus satisfying the acceptance criteria of Appendix A, "Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR," to Standard Review Plan (SRP) 15.1.5. Based on the foregoing considerations, the staff concludes that the potential radiological offsite consequences of a postulated MSLB, with a primary-to-secondary leakage less than the TS limit of 9.1 gpm for Braidwood, Unit 1, at the end of Cycle 5, are acceptable.

3.6 Risk Based Evaluation

The staff evaluated in the SE issued with Amendment No. 50, the acceptability of the SG IPC with specific attention focused on the issue of the level of risk created by the potential for SG tube rupture to lead to core damage. We reaffirm our prior evaluation on this matter and provide this section of our prior SE for the purpose of completeness.

The licensee's original IPC proposal of April 25, 1994, and its present proposal of June 20, 1994, contained two features not yet approved by the staff. One is the assumption that eddy current bobbin coil voltage signals have a POD equal to 1.0 for signal amplitudes over 3 volts. The other is that the TSPs will have limited motion during severe blowdown transients, including a postulated MSLB, which would allow credit for some constraining effects on cracks resulting from ODSCC in the TSP intersections with the SG tubes.

Because these two aspects of the licensee's analysis have not been accepted by the staff, the licensee also provided burst probabilities based on the assumptions of a POD value of 0.6 and taking no credit for TSP constraint (i.e., a free span analysis). These assumptions have been previously reviewed and accepted by the staff in draft NUREG-1477. With these conservative assumptions, the SG tube burst probability was estimated to be 3.1×10^{-2} at the beginning of the present fuel cycle and 9×10^{-2} at EOC as discussed in Section 3.3.2. These licensee estimates are for the probability that a single SG tube would burst if subjected to the effects of a secondary side depressurization following an MSLB event. The licensee also provided an estimate of the frequency of occurrence for rapid secondary side depressurizations that could possibly move the TSPs and allow bursting of axial tube ODSCC cracks in the TSP regions. This frequency is based on the occurrence of two feedwater line break (FWLB) events in 1370 reactor years of Westinghouse pressurized water reactor (PWR) operation, resulting in an estimated frequency of 1.8×10^{-3} FWLB/reactor-year. The licensee's analysis applied the same frequency to the MSLB. These break frequency values are higher than those previously used in staff analyses, but will be used for this review, pending further analysis of the underlying event data. It is noted that only large ruptures between the containment wall and the first blowdown restraining valve outside the containment will be significant for this risk estimate.

Using the licensee's values for initiating event frequencies and the induced SG tube rupture probabilities based on previously accepted calculational methods, the frequency of an induced single SG tube rupture is estimated to be:

$$(1.8 \times 10^{-3} + 1.8 \times 10^{-3})/\text{year} \times 3.1 \times 10^{-2} = 1.1 \times 10^{-4}/\text{year (BOC)}$$

and $$(1.8 \times 10^{-3} + 1.8 \times 10^{-3})/\text{year} \times 9 \times 10^{-2} = 3.2 \times 10^{-4}/\text{year (EOC)}$$

The licensee's analysis did not provide the estimated probability for failure to mitigate the combined effects of such events to prevent core damage. Previous staff analyses have used mitigation failure probabilities of 1×10^{-3}

(refer to NUREG-0844 and draft NUREG-1477) for the failure to mitigate the equivalent of the induced rupture of a single SG tube. Based on this value, the estimated frequency of core damage due to induced rupture of a single degraded SG tube at Braidwood, Unit 1, would vary from about $1x10^{-7}$/year at BOC to $3x10^{-7}$/year at EOC.

The staff's review also addressed the risk associated with the potential for inducing the rupture of multiple SG tubes since this was not addressed by the licensee. Accordingly, the staff estimated the probability of bursting two SG tubes and three SG tubes, based on information provided by the licensee. The breakdown of burst probability by individual eddy current indications allowed the staff to estimate that the conditional probability for inducing the rupture of two SG tubes at BOC is $6.5x10^{-4}$ and the probability for bursting three SG tubes is $9.8x10^{-6}$. The FWLB analysis contained in draft NUREG-1477 provides a basis for estimating the probability of mitigating a severe secondary side depressurization event with about three ruptured SG tubes. The estimated mitigation failure probability was $3.4x10^{-3}$. On that basis, we believe that the risk is dominated at BOC by the failure of a single SG tube.

We did not project the multiple SG tube failure values to EOC conditions, but we do not expect them to increase sufficiently to dominate the risk. (Because the single SG tube burst probability increased by about a factor of 3, we would not expect the burst probability at EOC for two and three SG tubes to increase by more than factors of 10 and 30, respectively, which would still leave the single SG tube risk dominant.) However, we believe that it is necessary to verify this evaluation. Depending upon the actual distribution of flaw indications, it is possible for the probability of multiple SG tube ruptures to reach levels of significant risk as the single SG tube rupture probability increases above previously estimated levels. Accordingly, the staff believes that the licensee should develop the methodology for evaluating the probability of multiple SG tube ruptures to be prepared for the mid-cycle inspection.

In summary, many probabilistic risk assessments for PWRs estimate total frequency of containment bypass releases on the order of 10^{-6} per reactor year and the estimated incremental risk for Braidwood, Unit 1, would not make it an outlier in terms of risk from the accident sequences discussed in this safety evaluation.

3.7 Leakage Monitoring and Other Administrative Controls

The licensee is maintaining the leakage monitoring and administrative control enhancements made following the October 1993 SG tube leak incident.

These safety enhancements included:

1. Lowering the alert and alarm setpoints on the main steam line and steam jet air ejector radiation monitors in both Units 1 and 2;

2. Making procedural changes to facilitate "quick counts" of chemistry samples to give rapid confirmation of SG leakage;

3. Increasing the chemistry sampling frequency to hourly when primary-to-secondary SG leakage is detected and then reducing the frequency to not less than daily when the SG leakage stabilizes;

4. Revising monitoring procedures to call for an hourly review of radiation monitor readings when SG leakage is detected;

5. Revising procedures to include the use of radiation monitor indications in the control room and the use of portable N-16 monitors to help ascertain SG leakage trends;

6. Upgrading training scenarios involving SG tube failures to include plant response data from an actual SG tube leak;

7. Revising SG leakage response procedures to better control and process contaminated secondary water resulting from a SG leakage event;

8. Verifying that steps in appropriate plant procedures continuously check for SG tube failure indications and do not use a "snap-shot" approach;

9. Revising control room surveillances to require that hourly trend readings of steam jet air ejector radiation monitor activity levels be reviewed on a daily basis.

Consistent with the IPC approach discussed in draft NUREG-1477, the licensee included the 150 gpd primary-to-secondary leakage limit in the Unit 1 TS for each SG. The licensee also added administrative SG leak trend limits to its procedures. With these administrative trend limits, plant shutdown is required in a 5-hour period if detectable leakage increases by 25 gpd per hour or more. For an increase above 100 gpd in an hour, plant shutdown is required in 4 hours. The 150 gpd leakage TS limit furnishes reasonable assurance that should a SG tube leak develop, it can be readily detected and the plant will be shut down before a tube rupture occurs. The 150 gpd value also provides for detection of SG leakage from a crack associated with the longest permissible freespan crack length. As discussed in the Westinghouse Report, WCAP-14046, the 150 gpd leakage corresponds to the leakage resulting from a 0.4 inch crack at nominal leak rates and a 0.6 inch long crack at 95% confidence level leak rates. This provides for plant shutdown prior to reaching critical crack lengths for postulated steam line break conditions at a SG leakage rate below the 95% confidence level and for the more restrictive three times normal operating pressure differential at less than nominal leak rates.

In summary, implementation of the above measures constitutes an acceptable defense-in-depth approach against tube failure and detection of flaws that would exceed steam line break leakage limits.

3.8 Tube Support Plate Deflections

To supplement the probabilistic arguments regarding SG tube burst, the licensee submitted TSP deflection analyses in WCAP-14046. These analyses indicate limited TSP movement, leading to the conclusion by the licensee that some degree of SG tube structural support was provided by the support plates during a postulated MSLB.

The TRANFLO code was used to predict the transient hydrodynamic loading on the TSPs caused by a postulated MSLB. These transient loads are in turn used to determine the structural response of the TSPs. In WCAP-14046, the licensee submitted generic TSP deflection analyses for Westinghouse Models D-3 and D-4 SG designs under various initial conditions. In its letter dated July 21, 1994, the licensee presented preliminary analyses based on the Braidwood Model D-4 SG design. These cases and their initiating conditions are summarized in Table 2.

Table 2

	WCAP (3) Case 1	WCAP Case 2	WCAP Case 3	Brwood Case 1	Brwood Case 2
SG Type	D-3	D-4	D-4	D-4	D-4
Mode (1)	HSB	Full Power	HSB, Excess Fdwater	HSB	HSB, Excess Fdwater
Water Level (2)	Normal	Normal	Low	Normal	Low

Notes:
(1) HSB = hot standby
Excess Fdwater = excess feedwater flow transient
(2) For D-4 design: Normal = 487" above the tube sheet
Low = 280" above the tube sheet, coinciding with the top TSP
(3) WCAP = WCAP-14046

The staff examined the results of the TRANFLO calculations to determine the reasonableness of the results generated by this code. The staff's previous review of TRANFLO concentrated on its calculation of the mass and energy release into a primary reactor containment during a postulated MSLB and did not include an evaluation of the ability of this code to accurately predict hydrodynamic loads on such internal structures as the TSPs.

The staff performed audit calculations of TSP hydrodynamic loads using the same Braidwood-specific model inputs and assumptions as those used by the licensee, attempting to match the modeling used in the two Braidwood-specific cases submitted by the licensee. The results of the staff's audit showed that the licensee's calculations are in general agreement with our independent audit both in the magnitude of the differential pressures predicted for the

TSPs and the variation of these differential pressures with respect to the initial SG water level. However, the staff audit calculation also showed indications of acoustic effects in the initial phases of the blowdown transients. Since the model used by the staff was intended to match as closely as possible the TRANFLO model used by the licensee, the noding and other modeling considerations were not optimized to investigate the acoustic effect. We note that the duration of the acoustic effect is extremely short. Accordingly, additional structural analysis may be needed to determine the impact of these potential acoustic loads on computed TSP displacements.

In this regard, the licensee stated that the TRANFLO model does not account for acoustic effects for the postulated MSLB transient. The licensee stated in its letter dated July 14, 1994, that acoustic effects were not expected to occur since the secondary side of the SG is in a two-phase condition. The compressibility of the mixture, the steam space existing during hot standby, and the flashing which would occur were also expected to restrict the magnitude of any acoustic effect. Until a more detailed assessment of TRANFLO is performed and additional review carried out regarding possible acoustic loading effects, the staff is unable to conclude that the TSP deflection analyses yield representative results.

The licensee is performing further plant-specific TSP deflection analyses. The staff believes that the licensee should address the possible acoustic effect due to a postulated MSLB. In that we have questions regarding the licensee's depiction of these hydrodynamic effects and the suitability of its proposed calculational methodology, the staff cannot, at this time, accept the results of the licensee's analysis of limited TSP deflections. We find, therefore, that the licensee's proposal to take credit under postulated accident conditions, for the structural support which might be provided by the TSPs for SG tubes subjected to the type of degradation identified as ODSCC, cannot be used as a licensing basis.

4.0 SUMMARY OF EVALUATION

4.1 Technical Summary

The staff has determined that Braidwood, Unit 1, may be safely operated past the 100 day operating limit imposed by Amendment No. 50. The licensee has applied a revised leak rate correlation, accepted by the staff, to determine the radiological consequences due to primary-to-secondary leakage induced by a postulated MSLB at end of the operating cycle. These calculations use conservative licensing basis assumptions and yield results that satisfy Standard Review Plan acceptance criteria and are well below the exposure guideline values of 10 CFR Part 100.

The staff has performed a deterministic structural assessment of steam generator tube integrity. The voltage based repair limit being applied considers the range of design basis events that could challenge the SG tube integrity and is consistent with the criteria of Regulatory Guide 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes." Under these deterministic criteria, the limiting consideration for Braidwood, Unit 1, is

to ensure that tubes can withstand 1.43 times the differential pressure imposed by a maximum possible MSLB, postulated to occur at the end of the present operating cycle. Based on the voltage distribution found during the last fuel cycle and the assumed voltage growth rate during the present fuel cycle, the staff found that the allowable voltage limit for the SG tubes (i.e., 4.5 volts) calculated on a deterministic basis, would be exceeded after the first 4.6 months of the present operating fuel cycle. Although all of the SG tubes would not satisfy the criterion of 1.43 times the MSLB differential pressure after this point in the fuel cycle, they would still have margin against burst. By limiting the operation period to March 1, 1995, a margin of about 1.15 against burst under MSLB differential pressure conditions will exist.

This 1.15 margin to burst is acceptable considering the additional assurance provided by compensatory leakage monitoring and mitigation measures, and the results of the risk evaluations performed by the staff. The deterministic SG tube integrity assessment makes several assumptions listed below which contribute to the conservatism of the staff's finding:

1. The 2560 psi differential pressure assumed to be imposed on the SG tubes in a conservative analysis is an upper bound of the pressure expected during an MSLB due to total depressurization of the secondary side of the faulted steam generator, while the reactor coolant system (RCS) pressure is simultaneously raised to the safety valve set points. An elevated RCS pressure would result only if operators did not carry out actions to terminate safety injection and to counter heat up of the RCS following the initial cooldown. Also, the pressure operated relief valves (PORVs) in the pressurizer are assumed to fail. The staff, therefore, expects that without these conservatisms, the actual differential pressure across the SG tubes would be below 2560 psi.

2. The voltage growth rate assumed in the deterministic analysis for the current fuel cycle is the maximum rate observed for the previous fuel cycle, rather than an average of the voltage growth rate. The licensee expects that improved molar ratio chemistry control and boric acid addition will reduce the present and future corrosion rates below those of the last fuel cycle. However, since any such improvement is difficult to predict, this assumption based on the previously observed maximum voltage growth rate, is used.

3. The burst pressure versus bobbin voltage correlation used is adjusted for lower SG tube structural properties. Some SG tubes could be assumed to be at this lower bound for material considerations, but there is no evidence that all of the SG tubes in the Braidwood, Unit 1, steam generators actually fall into this category.

4. The difference between the allowable voltage limit and the 1.0 volt IPC repair criterion includes an allowance for eddy current voltage measurement variability (i.e., the repeatability error) during the last SG inspection. This is equal to 20% of the initial voltage measurement.

In addition to the TS limits on the total and individual steam generator leakage, the licensee is maintaining the leakage monitoring and control enhancements made following the October 1993 SG tube leak incident. The more significant enhancements are listed below:

1. An administrative leak trend limit of 25 gpd per hour. Plant shutdown is required in a 5-hour period if detectable leakage increases by 25 gpd per hour or more. For an increase above 100 gpd in an hour, plant shutdown is required in 4 hours.

2. Primary-to-secondary leakage trending using control room indications are now part of normal plant operations.

3. The alert and alarm set points on main steam line and steam jet air ejector radiation monitors have been lowered.

4. Procedural changes to facilitate "quick counts" of chemistry samples to give rapid leak confirmation and increased chemistry sampling frequency when primary-to-secondary SG leakage is detected.

5. Leakage response procedural guidance has been upgraded to specifically direct the reactor operators to use radiation monitor indications in the control room and employ portable N-16 monitors to ascertain SG leakage trends.

6. Training scenarios have been incorporated addressing SG tube failures, including actual plant response data from the SG tube leak event of October 1993.

These measures continue to give added confidence in the margin of safety for the SG tubes to withstand normal operating and postulated accident conditions.

Finally, the staff assessed the level of risk created by the potential for the induced rupture of single and multiple SG tubes. The staff concludes that the estimated level of risk associated with the operation of Braidwood, Unit 1, would not be significant even for the full fuel cycle, assuming that the flaw growth rate continues at the same elevated rate seen in the previous fuel cycle.

Although the conservatisms of the deterministic structural assessment of SG tube integrity are not quantified, nor are the contributions to the margin of safety from the leakage monitoring enhancements cited above, the staff judges that these considerations provide adequate assurance of margin for the SG tubes to withstand the effects of a postulated MSLB. Further, the staff believes that the administrative actions taken by the licensee provide assurance that in the event SG tube integrity were compromised, the plant can safely mitigate the effects of SG leakage.

Based on the preceding evaluation, the staff requires the licensee to conduct a mid-cycle shutdown for steam generator inspection. This mid-cycle inspection should be initiated no later than March 1, 1995. To avoid delays

during the mid-cycle inspection, the staff requests that the licensee submit its plans for assessing the mid-cycle inspection data to ensure that it is consistent with what is expected (e.g., examining maximum and average growth rates, and comparing projections to actual distributions). The inspection data and repair results of the mid-cycle inspection, along with the licensee's safety assessment, will be reviewed by the staff prior to restart from the mid-cycle outage.

Based on the above considerations, the staff concludes that operation of Braidwood, Unit 1, until March 1, 1995, is acceptable and does not pose an undue risk to the public health and safety.

4.2 Approval of Technical Specification Revisions

The proposed change to TS Section 4.4.5.4.a(11) is acceptable in that it is editorial in nature; i.e, moving text from a footnote into the body of the section. The other proposed change to TS 4.4.5.4.a(11) is acceptable in that it replaces in Item 3, the licensee's prior calculated value of SG leakage under accident conditions at the end of 100 days (i.e., 26 gpm) with a revised value stated as less than 9.1 gpm at EOC. We find this revised TS value to be conservatively higher than the value of 6.8 gpm calculated by the staff at EOC. We also find that the editorial addition of the Westinghouse Report, WCAP-14046, to Item 3 of this TS section, is acceptable in that we agree with the licensee that there is a valid correlation between SG tube leakage and bobbin voltage amplitude, as discussed in Section 3.4.3. As stated in Section 3.5, we find that the potential radiological offsite consequences of a postulated MSLB are acceptable with the present TS limit of reactor coolant iodine-131 concentration of 0.35 microcuries per gram; this TS limit remains unchanged in accordance with the licensee's proposal in its letter dated August 18, 1994.

In summary, the net effect of these TS revisions is to incorporate a number of editorial changes, revise the calculated value of the SG leakage under accident conditions and removes from the TSs, for the remainder of the present fuel cycle, the operating limit on Braidwood, Unit 1.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, a reasonable effort was made to notify the Illinois State official of the proposed issuance of the amendment. Since no one was available at the time of notification, this matter will be discussed with the Illinois State official on August 19, 1994.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes a surveillance requirement. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative

occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (59 FR 35389). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: August 18, 1994

We further note that in connection with the preparation of this draft generic letter, an NRC staff member issued a memorandum containing a Differing Professional Opinion (DPO), dated July 13, 1994. The technical issues raised in this DPO are being evaluated in accordance with NRC policy. The subject DPO has subsequently been released to the Public Document Room (PDR) with permission of its author. Braidwood, Unit 1, was one of several nuclear power plants cited in this DPO. Both the CRGR and the ACRS have considered the subject DPO at their most recent meetings when reviewing the pending draft IPC generic letter.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:
Ramin R. Assa, Acting Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 54 to NPF-72
2. Safety Evaluation

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