Docket File

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

May 7, 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 NO. 50 TO FACILITY OPERATING LICENSE NPF-72, BRAIDWOOD STATION, UNIT 1 (TAC NO. M89166)

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 50 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 the request in your application dated April 25, 1994, as supplemented in your letters dated April 28, 1994, April 30, 1994, May 2, 1994, May 4, 1994, and May 6, 1994.

The amendment revises the TSs in Appendix A to the operating license by adding additional surveillance and operating requirements to Section 4.4.5.2, Section 4.4.5.4, and Section 4.4.5.5. These revisions incorporate the interim plugging criteria (IPC) and supplement the present surveillance requirements and acceptance criteria related to the steam generator (SG) tube repair criteria. The present TSs require that 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 thickness imperfection depth is equal to 40 percent or more of the nominal wall thickness. The revised TSs state that this definition of the tube repair criteria does not apply to the region of the tube at the intersection of the tubes with the tube support plates. Rather, at these intersections, the IPC will apply. The IPC is discussed in detail in the draft NRC report, NUREG-1477.

These revisions are effective only for 100 calendar days from the date of restart of Unit 1; any time below a T_{hot} temperature of 500°F will not be counted towards this time period. Since your submittal of April 25, 1994, proposed modifications to the IPC methodology not previously approved by the staff, the staff needs additional time to review these modifications. Accordingly, the staff suggested, and you agreed, to modify your original application by formally adopting the IPC methodology as contained in draft NUREG-1477. In your modified request, you also proposed to operate Unit 1 for a limited time to allow the staff to complete its review of your original proposal. The staff anticipates it will complete this effort within the time period cited above.

DFO1 CP/ 1/1

Mr. D. L. Farrar
Commonwealth Edison Company

Byron Station
Unit Nos. 1 and 2

cc:

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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. 50
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 April 25, 1994, as supplemented April 28, 1994, April 30, 1994, May 2, 1994, and May 6, 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. 50 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

James E. Dyer

James E. Dyer, 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: May 7, 1994

To support operations during this interim period, you also proposed additional changes to the Braidwood Station, Unit 1, Technical Specifications. Technical Specification 3.4.6.2, the operating leakage limiting condition for operation (LCO) is revised to reduce the permissible total primary reactor coolant system leakage to the secondary system from one gallon per minute (gpm) to 600 gallons per day (gpd) and to reduce the maximum permissible leakage through any one SG from 500 gpd to 150 gpd. The associated TS Bases, Section 3/4.4.5, are also revised to reflect the changes cited above. A footnote to TS 3.4.8 is also added which states that during the limited time of operation for Unit 1 cited above, the reactor coolant dose equivalent iodine-131 will be limited to 0.35 microcuries per gram of coolant.

While your original request was for an emergency license amendment, we found that there was not sufficient justification to grant it on that basis. However, we did find that you presented sufficient basis to process your amendment under exigent circumstances and have done so.

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,

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. to NPF-72
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:

Docket Files	NRC & Local PDRs	PDIII-2 r/f	J. Roe
J. Zwolinski	J. Dyer	R. Assa	C. Hawes
G. Hill(4)	OPA	OC/LFDCB	B. Clayton, RIII
OGC	D. Hagan	C. Grimes	ACRS(10)
E. Sullivan	E. Murphy	J. Strosnider	K. Karwoski
S. Long	A. El-Bassioni	K. Eccleston	*See Previous Concurrence

No legal objection

OFC	LA:PDIII-2	SPE:PDIII-2	PM:PDIII-2	BC:EMCB *	OGC	D:PDIII-2
NAME	CHAWES CMH	MDLYNCH	RASSA RA	JSTROSNIDER	for J Moore per telecon	JDYER JWD
DATE	5/7/94	05/07/94	5/7/94	5/7/94	05/07/94	5/7/94
COPY	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

ATTACHMENT TO LICENSE AMENDMENT NO. 50

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 4-13	3/4 4-13
3/4 4-14	3/4 4-14
-	3/4 4-14a
3/4 4-15	3/4 4-15
3/4 4-16	3/4 4-16
3/4 4-17	3/4 4-17
-	3/4 4-17a
-	3/4 4-17b
3/4 4-18	3/4 4-18
3/4 4-19	3/4 4-19
3/4 4-21	3/4 4-21
3/4 4-27	3/4 4-27
B 3/4 4-3	B 3/4 4-3
-	B 3/4 4-3a
B 3/4 4-4	B 3/4 4-4

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable steam generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube* Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. When applying the expectations of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by the sleeve are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

*When referring to a steam generator tube, the sleeve shall be considered a part of the tube if the tube has been repaired per Specification 4.4.5.4.a.10.

SURVEILLANCE REQUIREMENTS (Continued)

1) All tubes that previously had detectable tube wall penetrations greater than 20% that have not been plugged or sleeved in the affected area, and all tubes that previously had detectable sleeve wall penetrations that have not been plugged,

2) Tubes in those areas where experience has indicated potential problems,

3) At least 3% of the total number of sleeved tubes in all four steam generators or all of the sleeved tubes in the generator chosen for the inspection program, whichever is less. These inspections will include both the tube and the sleeve, and

4) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

d. For Unit 1 Cycle 5, implementation of the tube support plate interim plugging criteria limit requires 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 outer 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 principal indications can be characterized as ODSCC.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.

SURVEILLANCE REQUIREMENTS (Continued)

C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes or sleeves must exhibit significant (greater than 10% of wall thickness) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1) Reactor-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2c., or

   2) A seismic occurrence greater than the Operating Basis Earthquake, or

   3) A Condition IV loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   4) A Condition IV main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve;

3) Degraded Tube means a tube or sleeve containing unrepaired imperfections greater than or equal to 20% of the nominal tube or sleeve wall thickness caused by degradation;

4) % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing an unrepaired defect is defective;

6) Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair limit imperfection depth is equal to 40% of the nominal wall thickness. For Unit 1 Cycle 5, this definition does not apply to the region of the tube subject to the tube support plate interim plugging criteria limit, i.e., the tube support plate intersections. Specification 4.4.5.4.a.11 describes the repair limit for use within the tube support plate intersection of the tube;

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube that has been repaired by sleeving, the tube inspection shall include the sleeved portion of the tube, and

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 Criteria Limit is used in Unit 1 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. 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.

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*The plant specific guidelines used for all inspections shall be consistent with the eddy current guidelines in Appendix A of WCAP-13854.

SURVEILLANCE REQUIREMENTS (Continued)

3. The projected distribution of crack indications after 100 calendar days of operation from restart of Unit 1 Cycle 5, not counting any time below a T_{hot} temperature of 500°F, is verified to result in total primary to secondary leakage less than 26 gpm (includes operational and accident leakage). The basis for determining expected leak rates from the projected crack distribution is contained in Attachment A of CECo's letter dated April 30, 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 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, 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.

SURVEILLANCE REQUIREMENTS (Continued)

d. For Unit 1 Cycle 5, the results of inspection for all tubes in which the tube support plate interim plugging criteria limit has been applied shall be reported to the Commission pursuant to Specification 6.9.2 following completion of the steam generator tube inservice inspection and prior to Cycle 5 operation. The report shall include:

1. Listing of the applicable tubes,

2. Location (applicable intersections per tube) and extent of degradation (voltage), and

3. Projected Steam Line Break (MSLB) Leakage.

TABLE 4.4-1
MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	Yes
No. of Steam Generators per Unit	Four
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[1]

TABLE NOTATION

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N.A.	N.A.
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N.A.	N.A.
	C-3	Inspect all tubes in this S. G., plug or repair defective tubes and inspect 2S tubes in each other S. G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S. G.s are C-1	None	N.A.	N.A.
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug or repair defective tubes. Notification to NRC pursuant to §50.72(b)(2) of 10 CFR Part 50	N.A	N.A.

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 600 gallons per day total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 150 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, reduce the leakage rate to within limits within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Test pressures less than 2235 psig but greater than 350 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proporational to pressure differential to the one-half power.

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 Tavg greater than or equal to 500°F.

**For Unit 1 Cycle 5, the steam generators will be considered OPERABLE for the first 100 calendar days of operation with T_{hot} greater than 500°F. During that time, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.35 microCuries per gram.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator 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 steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired by sleeving. The technical bases for sleeving are described in Westinghouse report WCAP-13698 Rev. 1 and Babcock & Wilcox Topical Report BAW-2045PA Rev. 1.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the plugging or repair limit of 40% of the tube nominal wall thickness. If a sleeved tube is found to contain a through wall penetration in the sleeve of equal to or greater than 40% of the nominal wall thickness, the tube must be plugged. The 40% plugging limit for the sleeve is derived from Reg. Guide 1.121 analysis and utilizes a 20% allowance for eddy current uncertainty and additional degradation growth. Inservice inspection of sleeves is required to ensure RCS integrity. Sleeve inspection techniques are described in Westinghouse Report WCAP-13698 Rev. 1 and Babcock & Wilcox Topical Report BAW-2045PA Rev. 1. Steam Generator tube and sleeve inspections have demonstrated the capability to reliably detect degradation that has penetrated 20% of the pressure retaining portions of the tube or sleeve wall thickness. Commonwealth Edison will validate the adequacy of any system that is used for periodic inservice inspection of the sleeves and, as deemed appropriate, will upgrade testing methods as better methods are developed and validated for commercial use.

3/4.4.5 STEAM GENERATORS (continued)

For Unit 1 Cycle 5, tubes experiencing outer diameter stress corrosion cracking within the thickness of the tube support plates will be dispositioned in accordance with Specification 4.4.5.4.a.11.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.9.2 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 600 gpd for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 600 gpd limit is consistent with the assumptions used in the analysis of these accidents. The 150 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the Safety Injection flow will not be less than assumed in the safety analyses.

The 1 gpm leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, those valves should be tested periodically to ensure low-probability of gross failure.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NO. NPF-72

COMMONWEALTH EDISON COMPANY

BRAIDWOOD STATION, UNIT NO. 1

DOCKET NO. STN 50-456

1.0 INTRODUCTION

By letters dated April 25, 1994, as supplemented April 28, 1994, April 30, 1994, May 2, 1994, May 4, 1994, and May 6, 1994, Commonwealth Edison Company (CECo, the licensee), submitted a request to change the technical specifications for Braidwood Station, Unit 1. The requested amendment revises, in part, Technical Specifications 3.4.6.2, 3.4.8, 4.4.5.2, 4.4.5.4 and 4.4.5.5 and Bases 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 elevations. All of the proposed changes are applicable only for a limited time period in the fifth operating cycle.

The proposed voltage criteria pertains specifically to outside diameter stress corrosion cracking (ODSCC) flaws, and the proposed criterion: (1) permits flaws within the bounds of the tube support plate elevations with bobbin voltages less than or equal to 1.0 volt to remain in service; (2) permits flaws within the bounds of the tube support plate 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 tube support plate elevations with bobbin voltages greater than 2.7 volts to be plugged or repaired.

The staff is currently developing a generic interim position on voltage-based limits for ODSCC at tube support plate elevations. The staff has published several tentative conclusions regarding voltage-based plugging criteria in draft NUREG-1477. However, the staff is continuing to evaluate an acceptable generic position which takes into consideration public comments received on draft NUREG-1477, domestic operating experience under the voltage-based repair criteria, and additional data which has been made available from European nuclear power plants. The staff currently plans to document its final position in a generic letter with the disposition of public comments being documented in the final version of NUREG-1477. In the meantime, pending completion and issuance of the staff's generic position on the voltage-based interim plugging criteria (IPC), the staff is continuing to evaluate IPC proposals on a case-specific basis, as necessary, to ensure that there is adequate assurance of public health and safety.

2.0 BACKGROUND

By letter dated April 25, 1994, CECo requested an emergency amendment to modify the technical specifications to allow the usage of a voltage based plugging criteria for cycle 5 operations. Based on subsequent discussions between the licensee and the NRC staff, the licensee provided additional information and clarifications by letters dated April 30, May 2, May 4, and May 6, 1994.

The staff determined that the public should be given advance notice prior to issuing the amendment and that an emergency situation did not exist. Therefore, the staff chose to review the amendment request on an exigent basis. By letter dated April 30, 1994, CECo provided justification to support operation of Braidwood, Unit 1, for a limited time period based on reducing the allowable reactor coolant leakage rates and reactor coolant dose equivalent iodine-131 concentration from 1.0 to 0.35 microcuries per gram. This letter also contained a revised no significant hazards consideration. The staff, thereafter, published its finding of no significant hazards consideration in two local papers (Joliet News Herald and Morris Daily Herald) on May 3, 1994.

The tube repair limits included a 1.0 volt repair criterion for axially oriented ODSCC flaws confined to within the thickness of the tube support plate in lieu of the depth-based limit of 40%. In addition, the repair limits allowed bobbin indications between 1.0 and 2.7 volts are allowed to remain in service provided RPC inspection of these indications did not confirm the degradation to be present. The licensee's current proposal for Unit 1 is applicable for a limited time in Cycle 5 to permit the staff to complete its review of the licensee's original submittal.

The licensee's IPC proposal is similar to that reviewed and approved for several other plants and has been reviewed on a case-specific basis for Braidwood, Unit 1. Some of the features of this proposal are:

1. Calculation of the tube structural limit is based on maintaining a margin of safety of 1.43 against tube failure under postulated accident conditions vice maintaining a margin of safety of 3 against burst during normal operation.

2. A 1.0 volt IPC limit has been proposed.

3. The threshold for performing RPC examinations is that all flaw indications with bobbin voltages greater than 1.0 volt and less than or equal to 2.7 volts will be inspected by an RPC probe.

4. The methodology for calculating primary-to-secondary leakage from the steam generator tubes during a postulated main steam line break (MSLB) is in accordance with the methods described in draft NUREG-1477.

5. The diameter of the bobbin coil probe to be used in inspecting the steam generator tubes is 0.610 inches.

To evaluate the 1.0 volt IPC proposal for Braidwood, Unit 1, the staff considered not only the licensee's submittals but also operating experience from other nuclear power plants, foreign operating experience, and public comments received on draft NUREG-1477.

3.0 PROPOSED INTERIM PLUGGING CRITERIA

Braidwood, Unit 1, Technical Specifications 3.4.6.2; 3.4.8; 4.4.5.2; 4.4.5.4; 4.4.5.5; and Bases 3/4.4.5, are revised to specify the tube repair and leakage criteria for ODSCC at the tube support plate elevations. The tube repair and leakage criteria, based on the analysis in draft WCAP-14046 (Westinghouse Proprietary Class 2), and documentation contained in Electric Power Research Institute (EPRI) Report TR-100407, Revision 1, "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates" are:

a. An eddy current examination using a bobbin probe will be performed for 100% of the hot leg steam generator tube support plate intersections and 100% of the cold leg intersections down to the lowest cold leg tube support plate with ODSCC indications.

b. Degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage less than or equal to 1.0 volts will be allowed to remain in service.

c. Degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage greater than 1.0 volts will be repaired or plugged except as noted in (d) below.

d. Indications of potential degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage greater than 1.0 volts but less than or equal to 2.7 volts may remain in service if a RPC probe inspection does not detect degradation. Indications of ODSCC degradation with a bobbin voltage greater than 2.7 volts will be plugged or repaired.

4.0 EVALUATION

4.1 Inspection Issues

In support of the proposed interim repair limit, the licensee stated in its letter dated May 6, 1994, that it used guidelines that were in accordance with the standardized industry guidelines. The plant specific guidelines which the licensee used are consistent with those contained in Appendix A of WCAP-13854, which was referenced in Amendment Nos. 111 and 105 for Catawba, Units 1 and 2, respectively, and issued on December 16, 1993. These guidelines contain, in part, the bobbin specifications, calibration requirements, specific acquisition and analyses criteria, and flaw recording guidelines which should be used in the inspection of the steam generator tubes.

The inspection guidelines, the licensee's submittals, and the IPC methodology contain requirements, in part, to: (1) record all indications regardless of voltage (this is required to assess postulated MSLB leakage and probability of burst); (2) perform RPC inspections of 100 tubes, including tubes with bobbin dent voltages exceeding 5 volts and also including tube support plate intersections with artifact indications or indications with unusual phase angles; (3) perform RPC examinations of all tubes with bobbin voltages in excess of 1.0 volt; and (4) inform the staff prior to Unit 1, Cycle 5 operation of any unexpected RPC findings relative to the assumed characteristics of the flaws at the tube support plates which includes any detectable circumferential indications or detectable indications extending outside the thickness of the tube support plate.

While the licensee's letter dated May 2, 1994, confirmed that it had performed RPC inspections of all tubes with bobbin dent voltages exceeding 5 volts, which is consistent with the methodology in draft NUREG-1477, the licensee subsequently stated in the attachment to its letter dated May 6, 1994, that RPC inspections were performed on a representative sample of intersections rather than all intersections with dent indications exceeding 5 volts and with artifact indications (e.g., copper signals, mix residuals) exceeding 1 volt, with none found to contain flaw-like indications. The staff notes that draft NUREG-1477, which the licensee has committed to follow, recommends that all intersections with greater than 5 volt dents and all intersections where mix residuals that could cause a 1 volt flaw indication to be missed or mis-read, should be inspected with RPC. However, the staff agrees with the licensee that the RPC inspections implemented at Braidwood, Unit 1, are acceptable given that the potential for missed indications are accounted for in a conservative manner in the probability of burst and leak rate analyses described in Sections 4.3 and 4.4 of this safety evaluation.

With respect to the quality assurance (QA) review conducted by the licensee for implementation of the IPC, the licensee stated in its letter dated May 6, 1994, that all safety-related work associated with the steam generator repair was conducted in accordance with the QA Program requirements identified in Appendix B to 10 CFR Part 50. This was accomplished by the licensee's onsite quality verification group which sampled various activities both throughout the current refueling outage as well as during the Unit 1 forced outaged in October 1993. This group also conducted field monitoring reviews including reviews of the eddy current testing, tube plugging, and plug verification. On the basis that the Unit 1 steam generator repair was inspected by the licensee in accordance with Appendix B to 10 CFR Part 50, we find that implementation of the IPC satisifies the required quality assurance criteria.

The staff notes that there are several outstanding technical issues with respect to the inspection guidelines, as documented in previously issued NRC documents (e.g., in draft NUREG-1477) which must be resolved prior to adopting generic voltage limits in the IPC methodology. However, the staff concludes that the inspection guidelines proposed by the licensee for the Braidwood, Unit 1, including the plant specific 1.0 volt IPC are acceptable since the calibration, recording, and analysis requirements were consistent with the guidelines in Appendix A to WCAP-13854 as cited above.

4.2 Tube Integrity Issues

The purpose of the previous technical specification tube repair limits and the proposed IPC is to ensure that 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 14, 15, 31 and 32 of 10 CFR Part 50, Appendix A. Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the steam generator tubing. Leakage integrity refers to limiting primary-to-secondary leakage to within acceptable limits. The traditional strategy for accomplishing these objectives has been to establish a minimum wall thickness requirement in accordance with the structural criteria of Regulatory Guide 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 Regulatory Guide 1.121, to arrive at a depth-based repair limit. Enforcement of a minimum wall thickness requirement would implicitly serve 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 technical specifications to ensure timely plant shutdown before adequate structural integrity and leakage integrity of an affected tube is impaired.

The proposed interim tube repair limits for Braidwood, Unit 1, consist of voltage amplitude criteria rather than the traditional depth-based criteria. Thus, the repair criterion represents a departure from the past practice of explicitly enforcing a minimum wall thickness requirement.

The industry-wide database from the pulled tube examinations show that for bobbin indications exceeding 1.0 volts (i.e., the proposed IPC repair limit) maximum crack depths range 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 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 tubes which will be allowed to remain in service under the proposed IPC may have or develop through-wall or near through-wall crack penetrations during the upcoming cycle, thus creating the potential for leakage during normal operation and postulated MSLB accidents. The staff's evaluation of the proposed repair criteria from a structural and leakage integrity standpoint is provided in Sections 4.3 and 4.4 of this evaluation.

4.3 Structural Integrity

The licensee has proposed in Attachment F of its submittal dated April 25, 1994, a burst pressure/bobbin voltage correlation to demonstrate that bobbin indications satisfying the 1.0 volt interim repair criterion would retain adequate structural margins, consistent with the criteria of Regulatory Guide 1.121. The correlation was developed from both pulled tube data from other PWR plants (using pre-pull bobbin voltages) and laboratory tube specimens containing ODSCC flaws. The bobbin voltage data used to construct the burst pressure/bobbin voltage correlation were normalized to be consistent with the calibration standard voltage set-ups and voltage measurement procedures at Braidwood, Unit 1. During the current outage, the licensee pulled portions of 4 tubes containing 13 tube support plate intersections from the Braidwood, Unit 1, steam generators for leak and burst testing and destructive metallographic examination. These tests and examinations are being performed to confirm the expected axial ODSCC morphology and to enhance the supporting data bases for the burst and leakage correlations.

The interim voltage repair criteria approved by the staff to date for other plants have been set deterministically to ensure that tubes will retain adequate structural integrity during the full range of normal, transient, and postulated accident conditions with adequate allowance for eddy current test uncertainty and flaw growth during the next operating cycle. Because ODSCC is confined within the thickness of the tube support plates during normal operation, the staff has concluded that the structural constraint provided by the tube support plates ensures that all tubes with ODSCC will retain a margin of 3 with respect to burst under normal operating conditions, consistent with the criteria of Regulatory Guide 1.121, irrespective of the voltage amplitude of the indication. For a postulated MSLB accident, however, the tube support plates may displace axially during blowdown such that the ODSCC affected portion of tubing may no longer be fully constrained by the tube support plates. Thus, it is appropriate to consider the ODSCC affected regions of the tubes as free standing tubes for the purpose of assessing burst integrity under MSLB conditions.

The allowable end of cycle (EOC) voltage which ensures a margin of 1.43 with respect to burst under MSLB conditions (per Regulatory Guide 1.121) is based on the lower 95% confidence interval of the burst pressure versus voltage correlation contained in Attachment F of the licensee's submittal dated April 25, 1994, and is equal to 4.5 volts for the 3/4-inch diameter tubing used in the Braidwood, Unit 1, steam generators. The difference between the 4.5 volt allowable EOC voltage and the proposed 1 volt repair criterion represents an allowance for voltage growth during the cycle and for eddy current voltage measurement variability (i.e., the repeatability error). For Braidwood, Unit 1, maximum EOC voltages for Cycle 5 are expected to be dominated by the contribution of voltage growth during the course of the upcoming fuel cycle based on the voltage growth rates observed during the previous cycle which ranged up to a maximum of 9.8 volts per cycle. The licensee expects that an improved molar ratio (i.e., the ratio of sodium to chloride) chemistry control and boric acid addition will reduce future corrosion rates. The staff notes that any such improvement is difficult to quantify and is somewhat problematic. For this reason it is appropriate to assume that growth rates will be similarly distributed as those observed in the previous cycle. The contribution of the initial beginning of cycle (BOC) voltage value to the maximum EOC voltage is relatively small for 1 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 reading (i.e., equal to or less than 0.2 volts). Thus, voltages are expected to range to a maximum of between 9.8 and 11 volts by the end of the upcoming operating cycle, well in excess of the allowable EOC voltage limit of 4.5 volts. The allowable voltage limit could be satisfied by limiting operation of Unit 1 during the forthcoming fuel cycle to 4.6 months, at which time the steam generators would need to be reinspected to determine whether the accelerated voltage growth rate during the last fuel cycle was continuing.

The proposed 1 volt repair criterion is applicable to all bobbin indications confirmed by RPC or which have not been RPC inspected. The licensee is also proposing a 2.7 volt repair criterion applicable to bobbin indication locations which have been RPC inspected but for which the RPC failed to confirm the bobbin indication. This 2.7 volt criterion is the same as that recently approved for Catawba which also utilizes 3/4-inch diameter steam generator tubes. The numerical value of this criterion was originally determined on the basis of the allowable EOC voltage, allowing for the average growth rate observed at Catawba and the upper 95% confidence estimate for voltage variability. However, the measured growth rates at Braidwood, Unit 1, are significantly higher than at Catawba. Accommodating the average growth rate in the most limiting Braidwood, Unit 1, steam generator would lead to a voltage-based criterion of 2.2 volts as the upper limit for tubes which may be left in service if detected by bobbin inspection but not confirmed by the RPC. The staff believes that this discrepancy in the upper voltage limit allowed to remain in service, if not confirmed by RPC, between the Catawba and Braidwood, Unit 1, approaches is not significant on the basis that ODSCC which is not detectable by RPC is believed to be less likely to affect the tube structural or leakage integrity during the next operating cycle than ODSCC which is detectable by both bobbin and RPC inspections. In addition, the burst and leakage potential for bobbin indications accepted for continued service under the 2.7 volt criterion have been directly considered in the probability of burst and leakage assessments described below, with no credit given to the fact that RPC failed to confirm the indications. Based on these two considerations, the staff finds that the use of the 2.7 volts criterion for Braidwood, Unit 1, as an upper limit for tubes which may be left in service, is acceptable.

The licensee has performed a probabilistic analysis of the potential for 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% confidence interval used to determine the maximum allowable EOC voltage. Given the large numbers of indications being accepted for continued service with the proposed 1 volt criterion, the probabilistic analysis ensures that the use of the 95% confidence interval value in lieu of the 99% or 99.9% values does not lead to a significant likelihood of tube rupture given an MSLB.

2. The deterministic assessment ignores the burst and leakage potential of bobbin indications between 1 volt and 2.7 volts for which RPC failed to confirm the indication. The probabilistic assessment considers the burst potential of these indications with no credit given to the fact that RPC failed to confirm 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 should address the burst potential of indications missed by the data analysts.

The licensee's analysis applied the results of Monte Carlo simulations of the voltage measurement uncertainty distribution and the voltage growth rate distribution to the indications assumed to exist at the beginning of the forthcoming fuel cycle to arrive at a projected end of cycle distribution of indications. For each projected EOC indication voltage, the distribution of burst pressures as a function of bobbin voltage was sampled by a Monte Carlo technique to yield a frequency density function of burst pressures for that indication. Consistent with the approach recommended in draft NUREG-1477, the licensee's probabilistic analysis incorporates a probability of detection (POD) adjustment to the assumed BOC voltage distribution, 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. Identified indications exceeding the voltage criteria are subtracted from this distribution since the affected tubes will have been repaired prior to restart of Unit 1 to yield the assumed BOC distribution.

This approach 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 ODSCC at tube support plates is that maximum EOC 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 leads to a calculated conditional probability of rupture of 3.1E-2 at BOC and 9E-2 at EOC for a postulated MSLB. 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 EPRI performance demonstration program and the licensee's reanalysis of previous eddy current inspection data. The licensee anticipates obtaining additional insights on the detectability of large amplitude indications based on further study of the results of the recently completed steam generator 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 of 2E-3 at BOC and 5E-3 at EOC, which are less than the conditional probability estimates considered in the NUREG-0844 analysis of risk.

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 as large as 3 to 10 volts. However, the same logic is true for voltage indications in the 1 to 3 volt range. Moreover, 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. In the meantime, 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 4.5 of this evaluation.

The licensee has also performed a probability of burst analysis that considers the constraining influence of the tube support plates in mitigating the potential for burst during a postulated MSLB and which, therefore, is considered by the licensee to provide a more realistic analysis than that described above. This analysis includes an assessment of predicted support plate displacements during an MSLB and the portion of an ODSCC defect which may be exposed (i.e., unconstrained) as a result of these displacements. 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 tube support plate 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 1.7E-5 in the event of an MSLB. Since the licensee's analysis is under NRC staff review at this time, the conclusions of the staff's evaluation do not rely on the results of the licensee's analysis which takes credit for limited displacements of the tube support plates.

4.4 Leakage Integrity

4.4.1 Normal Operational Leakage

An important implication of voltage-based steam generator tube repair criteria is that the criteria may permit tubes with up to 100% through-wall cracks to remain in service. Thus, the leakage integrity of these tubes in addition to their structural integrity, must be assessed. Adequate leakage integrity during normal operation is implicitly assured by the allowable limits on the operational leak rate in the plant technical specifications. The licensee has proposed a change to the Braidwood, Unit 1, Technical Specifications which incorporates a more restrictive operational limit than that previously authorized. Specifically, the licensee has proposed to limit the primary-to-secondary leak rate through any one steam generator to 150 gallons per day (gpd) and to limit the total primary-to-secondary leak rate through all steam generators to 600 gpd. The current technical specification limits are 500 gpd through any one steam generator and 1.0 gallon per minute (gpm) (1440 gpd) through all four steam generators.

Since the proposed leak rate limits are more restrictive than the existing limits, they thereby 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 tube support plates or unexpectedly high crack growth rates leading to normal operational leakage. On this basis, the staff finds the proposed change to the primary-to-secondary leak rate limits to be acceptable.

4.4.2 Accident Leakage

The licensee has proposed a model for calculating the total steam generator tube leakage from the faulted steam generator during a postulated MSLB. The model consists of two major components; 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 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 has considered six different functional forms of the statistical model consistent with the approach recommended in draft NUREG-1477. The staff previously required that all six functional forms of the model be considered in the leak rate estimates performed in support of the voltage-based repair criteria approved for several plants implementing the IPC. 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 total leak rate estimate by a factor of 2 to 4. The staff has now concluded that use of a single functional form, the log-logistic, is acceptable for the purposes of assessing MSLB-induced leakage. The staff believes that any non-conservatism associated with the use of a log-logistic model as compared to the other functional forms is small compared to the conservatisms inherent in the existing methodology for estimating the radiological consequences of leakage induced by a postulated MSLB.

Regarding the conditional leak rate model, the licensee has proposed a correlation between leak rate and voltage based on a linear regression fit of the logarithms of the corresponding leak rate and voltage data. The staff has previously concluded (see draft NUREG-1477) that no proven relationship between leak rate and voltage had been demonstrated. However, the staff's preliminary review of the technical justification for the licensee's proposed model indicates that the licensee has in fact demonstrated a correlation between leak rate and voltage. The staff has not completed its review, however, of the proposed methodology for using the conditional leak rate model in conjunction with the POL model to calculate total leak rate in the faulted steam generator.

Pending completion of the staff's review of the licensee's proposed leak rate calculation methodology, the licensee has performed a more conservative leak rate analysis based on the methodology defined in draft NUREG-1477. This analysis includes consideration of all bobbin indications accepted for continued service. In addition, this analysis considers the leakage potential of non-detected bobbin indications assuming a POD of 0.6 independent of voltage. The leak rate calculated by the licensee in the most limiting Unit 1 steam generator was 18.9 gpm for an MSLB assumed to occur at BOC and 47.2 gpm for an MSLB assumed to occur at EOC.

The licensee stated in its submittal dated April 25, 1994, that the allowable leak rate was 9.1 gpm in the faulted steam generator consistent with maintaining the radiological consequences of a radiological 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 Standard Review Plan, and the utilization of thyroid dose conversion factors consistent with ICRP-30. To support plant restart, the licensee has committed in its letter dated April 30, 1994, to administratively control the Unit 1 reactor coolant dose equivalent iodine-131 activity to 0.35 microcuries per gram of coolant. This reduction in the reactor coolant system iodine concentration increases the allowable leak rate to 26 gpm. The potential for primary-to-secondary leakage following a postulated MSLB is assumed to increase in a linear manner over time. Based on an assumed operating cycle of 1.15 years, the most limiting Braidwood, Unit 1, steam generator can be operated for 3.4 months during which time MSLB-induced leakage is calculated on a conservative basis to be less than the allowable 26 gpm. Our detailed review of the radiological consequences of primary-to-secondary leakage in the event of an MSLB is contained in Section 4.6 of this evaluation.

4.5 Risk Based Evaluation

The staff reviewed the adequacy of the steam generator IPC focused on the issue of the level of risk created by the potential for steam generator tube rupture to lead to core damage.

The licensee's IPC proposal contains two features not yet approved by the staff. One is the assumption that eddy current bobbin coil voltage signals have a probability of detection (POD) equal to 1.0 for signal amplitudes over 3 volts. The other is that the tube support plates (TSPs) will have limited motion during severe blowdown transients, including a postulated MSLB, which would allow credit for some remaining constraining effects on cracks resulting from ODSCC in the TSP intersections with the tubes.

Because these two aspects of the analysis have not been accepted by the staff, the licensee also provided burst probabilities based on the assumptions of a value for POD of 0.6 and no tube support plate constraint (i.e., a free span analysis). These assumptions have been previously reviewed and accepted by the staff in draft NUREG-1477. These estimated values of tube burst probability were 3.1×10^{-2} at the beginning of the forthcoming fuel cycle and 9×10^{-2} by the end of the forthcoming fuel cycle. These licensee estimates are for the probability that a single 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 tube support plates and allow bursting of axial tube ODSCC cracks in the TSP regions. That frequency is based on the occurrence of two feedwater line break (FWLB) events in 1370 reactor years of Westinghouse 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 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 steam generator tube rupture probabilities based on previously accepted calculational methods, the frequency of an induced single steam generator tube rupture is estimated to be:

$$(1.8 \times 10^{-3} + 1.8 \times 10^{-3})/\text{year} \times 3.14 \times 10^{-2} = 1.1 \times 10^{-4}/\text{year} \quad \text{(BOC)}$$

and

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

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 tube. Based on this value, the estimated frequency of core damage due to induced rupture of a single degraded tube at Braidwood, Unit 1, would vary from about 1×10^{-7}/year at BOC to 3×10^{-7}/year at EOC.

The staff's review also addressed the risk associated with the potential for inducing the rupture of multiple tubes, which was not addressed by the licensee. The staff estimated that the probability of bursting two tubes and three tubes, based on information provided by the licensee. The breakdown of burst probability by individual eddy current indication allowed us to estimate that the conditional probability for inducing the rupture of two tubes at BOC is 6.5×10^{-4} and the probability for bursting three tubes is 9.8×10^{-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 tubes. The estimated mitigation failure probability was 3.4×10^{-3}. On that basis, we believe that the risk is dominated at BOC by the failure of a single tube.

We did not project the multiple tube failure values to EOC conditions, but do not expect them to increase sufficiently to dominate the risk. (Because the single tube burst probability increased by about a factor of 3, we would not expect the burst probability at EOC for two and three tubes to increase by more than factors of 10 and 30, respectively, which would still leave the single tube risk dominant.) However, we believe that it is necessary to verify this evaluation by requesting the licensee to provide estimates for the probability of rupturing multiple tubes. Depending upon the actual distribution of flaw indications, it is possible for the probability of multiple tube ruptures to reach levels significant to risk, as the single tube rupture probability is allowed to increase above previously estimated levels.

The estimated incremental frequency of core damage can also be considered to be the incremental frequency for containment bypass release. Therefore, the acceptability of these results were evaluated on that basis. Many probability risk assessments (PRAs) for pressurized water reactors (PWRs) have produced total frequencies of containment bypass releases in the low 10^{-6}/year range, which the staff has found acceptable. On that basis, the estimated increment for Braidwood, Unit 1, appears to be in the acceptable range. However, it should be noted that the Byron Individual Plant Examination (IPE), which has just been submitted and should be representative of the Braidwood Station, estimates a bypass release frequency in the 10^{-8}/year range. On that basis, and assuming that the Braidwood IPE will provide a similar estimate, our estimated increment to the bypass release frequency due to induced steam generator tube ruptures which could potentially occur using the proposed IPC, would be numerically dominant. However, the Byron IPE has not yet been reviewed by the staff, so the basis for the low value of the estimated frequency of bypass releases has not yet been verified by the staff.

On the basis of the risk assessment described above, the staff finds that the risk associated with the restart of Braidwood, Unit 1, falls in the acceptable range.

4.6 Radiological Consequences

In its submittal dated April 25, 1994, and as supplemented by its letter dated April 30, 1994, the licensee stated that it had calculated the permissible primary-to-secondary leakage during a postulated MSLB event based on maintaining the calculated dose rates at the site boundary to a small fraction of the guideline values in 10 CFR Part 100. Specifically, in its submittal of April 30, 1994, the licensee stated that based on its analysis, Braidwood, Unit 1, can operate for a limited period of time, if its technical specification dose equivalent iodine-131 concentration is reduced from 1.0 to 0.35 microcuries per gram of coolant. In its calculations, the licensee used the constant value of leakage, independent of bobbin voltage, contained in draft NUREG 1477 to develop predicted leak rates at BOC and EOC that can be expected during a postulated MSLB event. The licensee concluded that a post-MSLB maximum permissible leakage rate is 26 gpm in the limiting steam generator. Using its estimated BOC and EOC primary-to-secondary leakage rates and assuming a linear growth in these rates during plant operation, the licensee estimated a permissible operating period of 3.4 months.

The licensee's evaluation assumed both a pre-existing iodine spike and an event-generated iodine spike. In its evaluation, the licensee calculated site boundary thyroid doses and concluded that the event-generated spike case is limiting, and that the acceptance criteria for a MSLB with an assumed event-generated spike would be satisfied for a projected post-MSLB primary-to-secondary leak rate of 26 gpm as discussed above.

The staff has independently calculated the radiological consequences at the site boundary of a postulated MSLB assuming the licensing basis value for X/Q which is 5.6×10^{-4} sec/m^3. The staff used the dose conversion factors for iodine isotopes set forth in ICRP-30 as well as the breathing rates set forth in Regulatory Guide 1.4.

Table 1 presents the staff calculated thyroid doses for both the pre-existing iodine spike case and the event-generated spike case.

TABLE 1

[26 gpm Primary-to-Secondary Leak Rate]

CALCULATION TYPE	THYROID DOSE (REM) PRE-ACCIDENT IODINE SPIKE CASE	THYROID DOSE (REM) EVENT-GENERATED IODINE SPIKE	WHOLE-BODY DOSE (REM)
EAB (2 HOUR)	196	28	$\leq$0.1

As can be seen from this table, the calculated thyroid doses are within the exposure guideline values in 10 CFR Part 100 for the pre-existing iodine spike case, and thus, satisfy the acceptance criteria in Appendix A, "Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR," in Standard Review Plan 15.1.5. Similarly, the calculated thyroid dose for the event-generated iodine spike case are also shown in this table and is a small fraction of the exposure guideline values in 10 CFR Part 100, and thereby satisfy the acceptance criteria of SRP 15.1.5, Appendix A.

Accordingly, the staff concludes that the radiological consequences of a postulated MSLB resulting in a release outside containment for Braidwood, Unit 1, at 3.4 months are acceptable. This conclusion is based on a calculated maximum permissible post-MSLB primary-to-secondary leak rate of 26 gpm.

4.7 Systems Operational Measures

In October 1993, Braidwood, Unit 1, experienced a steam generator (SG) tube leak of about 300 gpd and the unit was shut down well before the technical specification leakage limit was reached. This event prompted leakage prevention and control enhancements outlined in Attachment B of the licensee's submittal dated April 25, 1994. The safety enhancements included: (1) lowered alert and alarm set points on main steam line and steam jet air ejector radiation monitors in both Units 1 and 2; (2) procedural changes to facilitate "quick counts" of chemistry samples to give rapid leak confirmation; (3) increasing the chemistry sampling frequency to hourly when primary-to-secondary leakage is detected and then reduced in frequency to at least daily when the leakage stabilizes; (4) additional monitoring procedures calling for hourly review of radiation monitor readings when leakage is detected; and (5) administrative primary-to-secondary leakage limits of 150 gpd maximum and a maximum increase of 25 gpd in a one-hour period. Use of radiation monitor indications in the control room, and portable N-16 monitors are included in the procedural guidance to help ascertain leakage trends. These measures provided additional assurance in the margin of safety for the tubes to withstand normal operating and postulated accident conditions as discussed in the safety evaluation issued with License Amendment No. 43 to Unit 1, dated December 16, 1993.

Consistent with the interim plugging criteria (IPC) approach discussed in draft NUREG-1477, the licensee's license amendment request contains a proposal to include the 150 gpd primary-to-secondary leakage limit in the Unit 1, technical specifications. In supplements to the original application, the licensee proposed additional administrative controls, including: training scenario updates; changes to leakage response procedures to control and process contaminated secondary water resulting from a leak event; and verification that appropriate plant procedures continuously check for steam generator tube failure indications as opposed to using a "snap-shot" approach. The licensee has also proposed to institute a change in the control room surveillances to require that hourly readings of steam jet air ejector radiation monitor activity levels be trended and reviewed on a daily basis.

The staff concerns for operation of Braidwood, Unit 1, for an interim period center on maintaining and strengthening where possible, the compensatory measures instituted following the October 1993 tube leak. These steps provide added assurance that Braidwood, Unit 1, can avoid a challenge from a steam generator tube failure, and if such an event occurs, to safely handle the resulting situation. In this regard, the licensee has proposed a number of additional measures to address our short-term concerns. These are: (1) minimization of secondary contamination in the event of a primary-to-secondary leak; (2) training related to operator action in the event of a steam generator tube failure and the expected indications in such situations; (3) the procedural approach used for steam generator tube failure diagnosis; (4) leakage trending during normal plant operation; and (5) justification of the 150 gpd leakage limit.

The licensee is maintaining the previous changes made following the October 1993, tube leak incident in Unit 1, and is instituting further steps which are cited above. Primary-to-secondary leakage trending using control room indications will become a part of normal plant operations. Additionally, training scenarios addressing potential steam generator tube failures are being further upgraded to include actual tube leak plant data. In addition to including the 150 gpd leakage limit in the Unit 1 Technical Specifications, the licensee is adding administrative leak trend limits to the 25 gpd per hour limit already in place. These additional administrative measures will now require plant shutdown in a 5-hour period if detectable leakage increases by 25 gpd per hour or more over a threshold value of 50 gpd. For an increase in the total leakage from any one steam generator above 100 gpd in an hour, plant shutdown will be required in 4 hours. These additional steps will provide assurance that a developing leak situation is promptly detected and that the plant is placed in a safe condition prior to any serious challenge from a significant tube failure.

The licensee has justified the 150 gpd leakage limit for the existing Unit 1 steam generator conditions in that the limit follows the guidance in draft NUREG-1477, providing reasonable assurance that should a tube leak develop, it can be readily detected and the plant shut down before a tube rupture. The 150 gpd value provides for detection of leakage from a crack associated with the longest permissible freespan crack length. As explained in Attachment F of the licensee's submittal dated April 25, 1994, a 150 gpd leakage corresponds to the leakage resulting from a 0.4 inch long crack at nominal leak rates and a 0.6 inch long crack at leak rates corresponding to the 95% confidence level. This provides assurance that the plant will be shut down prior to reaching critical crack lengths for steam line break conditions at leakage below the 95% confidence level and for the more restrictive 3 times normal operating pressure differential at less than nominal leak rates. This leakage limit, as well as the leakage trend limit and the leakage trending surveillance procedures being instituted, constitute an acceptable defense-in-depth approach against tube failure and detection of flaws that would exceed steam line break leakage limits.

Based on the foregoing considerations, the staff finds that the administrative measures being instituted by the licensee reduce the potential for a tube failure and furnish added means to mitigate the consequences of tube failure if it occurs. The operational leakage limit of 150 gpd, and the administrative leakage trend limits furnish sufficient confidence in the margin of safety for tubes to withstand loads imposed during normal operation or postulated accidents. The licensee has maintained and further augmented other compensatory actions, notably in the areas of leakage monitoring and tube failure mitigation procedures. These steps provide added assurance that the plant can be operated on an interim basis pending further review of the amendment request.

5.0 SUMMARY OF EVALUATION

A probabilistic risk assessment has demonstrated that restart and continued operation of Braidwood, Unit 1, does not pose undue risk to public health and safety. However, the calculated conditional probability of tube rupture given an MSLB is relatively high compared to that for other PWRs. This is due, in part, to the relatively high crack rates observed at Braidwood, Unit 1, during the most recent operating cycle. This probabilistic risk calculation conservatively ignores any reinforcement against tube rupture which may be provided by the tube support plates and utilizes conservative assumptions regarding the performance of the inservice inspection in detecting large amplitude indications. However, when combined with the probabilities of postulated initiating events and the failure of the reactor operators to effectively mitigate a tube rupture event, the staff finds that the overall risk is small.

A conservative analysis demonstrates that the radiological consequences of the potential primary-to-secondary leakage during a postulated MSLB will be within acceptable limits, consistent with 10 CFR Part 100, during at least the initial 3.4 months of the operating cycle. This finding is subject to administratively controlling the primary coolant dose equivalent iodine-131 concentration to 0.35 microcuries per gram of coolant as has been committed to by the licensee. In addition, the staff concludes that ODSCC indications accepted for continued service in accordance with the proposed interim voltage repair criteria will retain adequate structural margin during this period of time. The proposed reductions in the technical specification operational leakage limits provide added assurance of tube structural and leakage integrity.

Based on the above considerations, the staff concludes that the proposed restart and operation of Braidwood Unit 1, for a limited time period implementing the proposed changes to the technical specifications incorporating the voltage-based repair criteria and the proposed change for more restrictive limits on allowable operational leakage are acceptable. We also find that restart of Braidwood, Unit 1, does not pose an undue risk to public health and safety. However, the staff concludes that Braidwood, Unit 1, should be shutdown for another steam generator inspection within 100 calendar days from restart unless otherwise authorized by the staff. Operation during this limited period below a T_{hot} temperature of 500°F does not count towards this limit. The staff selected this temperature limit based on data indicating accelerated ODSCC crack growth above this temperature.

The staff is currently evaluating analyses submitted by the licensee to support operation beyond this interim period. These assessments involve more realistic modeling of leakage induced by MSLB, eddy current flaw detection performance for large amplitude indications, and structural reinforcement of tubes provided by the tube support plates. To facilitate the staff's review of these three issues, the licensee will be requested to submit the available test and examination results for the four tube specimens which were pulled from the Braidwood, Unit 1, steam generators on an expedited schedule.

6.0 NEED FOR EXPEDITED ACTION

The Commission's regulations, 10 CFR 50.91, contain provisions for issuance of amendments when the usual 30-day public notice period can not be met. One type of special exception is an exigency. An exigency is a case where the staff and licensee need to act promptly, but failure to act promptly does not involve a plant shutdown, derating, or delay in startup. The exigency case usually represents an amendment involving a safety enhancement to the plant.

Under such circumstances, the Commission notifies the public in one of two ways: by issuing a Federal Register notice providing an opportunity for hearing and allowing at least two weeks for prior public comments, or by issuing a press release discussing the proposed changes, using the local media. In the case of the licensee's submittal dated April 25, 1994, the Commission used the second approach since the staff believed that the public must be given advance notice prior to issuing the amendment and also believed that an emergency situation does not exist. The amendment request was noticed in two local newspapers on May 3, 1994, at which time the staff proposed a no significant hazards consideration determination.

While the staff's review of certain portions of the licensee's proposal is not complete at this time as discussed in Sections 4.3 and 4.5, the staff has concluded on a conservative basis that Braidwood, Unit 1, can be restarted for 100 calendar days of Cycle 5 as discussed in Section 5.0 of this safety evaluation. This limited operating period has been chosen in order to permit the staff sufficient time to complete its review of certain portions of the April 25, 1994, proposal.

The licensee requested in Attachment A of its submittal dated April 25, 1994, that this license amendment request be processed on an emergency basis. As discussed above, this request is now to be treated as an exigent license amendment. The licensee's basis for expedited action is that restart of Braidwood, Unit 1, is needed in a timely manner prior to CECo's peak summer load, especially in light of the fact that several other CECo facilities have experienced forced outages. Further, the licensee stated that any additional delays in the restart of Unit 1 may leave it with inadequate owned reserves as it enters its peak electrical demand.

The licensee stated in Attachment A cited above that it was unable to predict, prior to the present Unit 1 refueling outage and subsequent steam generator tube inspection, the large number of tube defects attributable to ODSCC. The detailed chronology in Attachment A demonstrates the numerous consultations with the staff which lead the licensee to conclude by April 21, 1994, that it had to pursue on a conservative basis, the use of the interim plugging criteria (IPC) contained in draft NUREG-1477 and as recently reviewed and approved by the staff for several other PWRs. The earliest date that the licensee was able to submit its formal proposal pursuing this IPC strategy was April 25, 1994. Having completed all refueling work efforts shortly thereafter, the restart of Unit 1 is now dependent on staff approval of the pending license amendment request implementing the IPC.

On the basis that the licensee made timely application of its pending license amendment request as discussed above and that the licensee has a need to restart Unit 1 in a timely manner, the staff finds that pursuant to 10 CFR 50.91(a)(6), exigent circumstances exist. Further, having used local media to provide reasonable notice to the public in the area surrounding the licensee's facility in accordance with Section 50.91(a)(6)(i)(B), the licensee's request of April 25, 1994, for a license amendment is being issued under exigent circumstances.

There were no public comments in response to the notices published in the local news papers.

7.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Implementation of the IPC for Braidwood, Unit 1, Cycle 5 meets the requirements of Regulatory Guide (RG) 1.121 by demonstrating that tube leakage is acceptably low and tube burst is a highly improbable event either during normal operation or a postulated MSLB event.

Under accident conditions, significant margins exist against the possibility of free span tube burst even for voltage growth in excess of the 95% cumulative probability. For the largest confirmed indications left in service, the licensee project a voltage growth is 2.6 volts at the end of cycle, which would result in the voltage of the largest indication left in service below the 4.5 volts structural limit for a free span burst pressure of 1.43 times the steam line break pressure differential tube. Even at a 99% cumulative probability, the observed voltage growth is bounded by 2.7 volts and the structural limit is satisfied for the bobbin coil indications between 1.0 and 2.7 volts left in service.

In addition, the licensee's analyses indicate that there are additional margins over those relied upon by the staff which are provided by the following considerations:

- A demonstration of limited tube support plate (TSP) displacement was done which reduces the likelihood of a tube burst. Limited TSP displacement would also reduce leakage compared to that calculated assuming free span ODSCC tube cracks.

- The Electric Power Research Institute (EPRI) Performance Demonstration Program analyzed the performance of some 20 eddy current data analysts evaluating data from a unit with 3/4-inch inside diameter and 0.049-inch wall thickness tubes. This data demonstrated that there is a voltage dependence of the probability of detection (POD) and argues for a POD of greater than 0.6 for ODSCC indications larger than 1.0 volt.

- CECo's risk evaluation of the operability of Braidwood, Unit 1, Cycle 5 compared core damage frequency, with containment bypass, with and without the IPC applied at Braidwood Unit 1. 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 its individual plant evaluation (IPE). CECo concluded that operation with the requested IPC resulted in an insignificant increase due to a postulated MSLB with containment bypass sequence frequency.
- The licensee's evaluation presented above applies to a full cycle of operation. Because plant operations approved by the proposed amendment would be for a significantly shorter period, the probability of an accident is much less than that calculated for the full cycle.

Additionally, to support the restart and limited operation of Braidwood Unit 1, the reactor coolant system dose equivalent iodine-131 will be limited to 0.35 microcuries per gram of coolant.

Therefore, since implementation of the 1.0 volt IPC for a limited time period for Braidwood, Unit 1, Cycle 5 does not adversely affect steam generator tube integrity and would not exceed acceptable dose consequences for a worst case postulated accident, this amendment does not result in any increase in the probability or consequences of an accident previously evaluated within the Braidwood Updated Final Safety Analysis Report.

The steam generator tube IPC does not introduce any significant changes to the plant design basis. Use of the IPC criteria does not provide a mechanism which could result in an accident outside the tube support plate elevations. Moreover, no ODSCC is occurring outside the thickness of the tube support plates. Neither a single or multiple tube rupture event would be expected in a steam generator in which the IPC has been applied. The low probability of this has been independently confirmed by the staff. Since neither the design basis of the plant nor the potential tube failure modes have been changed, the staff finds that the implementation of the IPC will not create the possibility of a new or different kind of accident from any accident previously evaluated.

With respect to the third criterion (i.e., a significant reduction in a margin of safety), CECo will implement a maximum leakage rate limit of 150 gallons per day (gpd) through any one Unit 1 steam generator to help preclude the potential for excessive leakage during all plant conditions. The criterion for establishing operational leakage rate limits which require plant shutdown are based upon leak-before-break considerations to provide assurance that a leaking free span crack can be detected before a potential tube rupture during faulted plant conditions. The 150 gpd limit will provide for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible free span crack length. Since tube burst due to ODSCC is precluded during normal operation due to the proximity of the TSP to the tube and the potential exists for an ODSCC crack to become uncovered during MSLB conditions, the leakage from the maximum permissible crack must preclude tube burst at MSLB conditions. Thus, the 150 gpd limit provides assurance that the plant will be shut down prior to reaching critical crack lengths for MSLB conditions.

Even under the worst case postulated accident conditions, the occurrence of ODSCC at the TSP elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The distribution of crack indications at the TSP elevations are confirmed to result in acceptable primary-to-secondary leakage during all plant conditions for the limited time period of this license amendment. The radiological consequences resulting from the maximum leakage calculated on a conservative basis, which could occur under faulted conditions (i.e., 26 gpm), are not adversely impacted.

A postulated loss of coolant accident (LOCA) coincident with a safe shutdown earthquake (SSE), as required by General Design Criteria (GDC) 2 of Appendix A to 10 CFR Part 50, may cause a tube collapse in the steam generators. A number of tubes have been identified in the "wedge" locations of the Unit 1 TSPs to represent a potential for tube collapse during a LOCA + SSE event. Accordingly, these tubes have been excluded from application of the voltage-based criteria (i.e., the IPC) being implemented in Unit 1.

With respect to the considerations in RG 1.83, "Inservice Inspection of PWR Steam Generator Tubes," Revision 1, July 1975, implementation of the bobbin coil probe voltage-based IPC of 1.0 volt is supplemented by: enhanced eddy current inspection guidelines to provide consistency in voltage normalization; a 100% eddy current inspection sample size at the tube support plate elevations; and RPC inspection requirements for the larger indications left in service (i.e., bobbin voltage indications between 1.0 and 2.7 volts) so as to be able to characterize the principal mechanism of degradation as ODSCC.

On the basis that the proposed implementation of the interim plugging criteria as presented in draft NUREG-1477 does not involve a significant reduction in a margin of safety as discussed above, the staff finds that the proposed license amendment satisfies the third criteria of 10 CFR 50.92.

Based upon the above considerations, the staff concludes that the amendment meets the three criteria of 10 CFR 50.92. Therefore, the staff has made a final determination that the proposed amendment does not involve a significant hazards consideration.

8.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

9.0 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. The NRC staff has determined that the amendment involves 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding. 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.

10.0 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: May 7, 1994