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

February 3, 1995

Mr. J. E. Cross
Senior Vice President and
Chief Nuclear Officer
Nuclear Power Division
Duquesne Light Company
Post Office Box 4
Shippingport, PA 15077

SUBJECT: ISSUANCE OF AMENDMENT NO. 184 TO FACILITY OPERATING LICENSE NO. DPR-66 IN RESPONSE TO CHANGE REQUEST NO. 207 REGARDING INTERIM STEAM GENERATOR TUBE PLUGGING CRITERIA (TAC NO. M89991)

Dear Mr. Cross:

The Commission has issued the enclosed Amendment No. 184 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit No. 1, in response to your application dated July 29, 1994, as supplemented in a letter dated December 13, 1994.

This amendment revises Technical Specifications (TSs) 3/4.4.5 and 3.4.6.2 including associated Bases 3/4.4.5 and 3/4.4.6.2 to allow the implementation of steam generator tube interim plugging criteria (IPC) for the tube support plate elevations during operating cycle 11. The current TSs require that tubes with imperfections exceeding 40 percent of the nominal tube wall thickness be removed from service. The IPC will allow a test voltage-based criterion of 1.0 volts as determined by a bobbin probe inspection of the tubes. Voltages greater than 1.0 volt will be further examined using a pancake coil probe. Tubes showing flaw indications with a bobbin voltage greater than 3.6 volts will be plugged or repaired.

As a condition of our approval of your proposal, the NRC staff has concluded that you should remove six tube support plate intersections for destructive analysis during the outage which began January 2, 1995. The issue of tube pulls was discussed with your staff during a conference call December 21, 1994. Since that time, you have completed removing the required 6 intersections for analysis, thus meeting this requirement.

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

Sincerely,

Original signed by
Gordon E. Edison, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosures: 1. Amendment No.184 to DPR-66
2. Safety Evaluation

cc w/encls: See next page

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

Sincerely,

G E Edison

Gordon E. Edison, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosures: 1. Amendment No. 184 to DPR-66
2. Safety Evaluation

cc w/encls: See next page

J. E. Cross
Duquesne Light Company

Beaver Valley Power Station
Units 1 & 2

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

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.184
License No. DPR-66

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

   A. The application for amendment by Duquesne Light Company, et al. (the licensee) dated July 29, 1994, as supplemented in a letter dated December 13, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the 2Act) 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.

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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. DPR-66 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 184 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A E Edison for

Walter R. Butler, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 3, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 184

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following pages of Appendix A, Technical Specifications, with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

| Remove | Insert |
|---|---|
| 3/4 4-9 | 3/4 4-9 |
| 3/4 4-10 | 3/4 4-10 |
| 3/4 4-10a | 3/4 4-10a |
| 3/4 4-10b | 3/4 4-10b |
| 3/4 4-10c | 3/4 4-10c |
| 3/4 4-10d | 3/4 4-10d |
| 3/4 4-10e | 3/4 4-10e |
| --- | 3/4 4-10f |
| 3/4 4-13 | 3/4 4-13 |
| B 3/4 4-2a | B 3/4 4-2a |
| B 3/4 4-3g | B 3/4 4-3g |
| B 3/4 4-3h | B 3/4 4-3h |
| B 3/4 4-3i | B 3/4 4-3i |
| B 3/4 4-3j | B 3/4 4-3j |

SURVEILLANCE REQUIREMENTS (Continued)

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

3. At least 3 percent of the total number of sleeved tubes in all three steam generators. A sample size less than 3 percent is acceptable provided all the sleeved tubes in the steam generator(s) examined during the refueling outage are inspected. These inspections will include both the tube and the sleeve, and

4. A tube inspection pursuant to Specification 4.4.5.4.a.8. If any selected tube does not permit the passage of the eddy current probe for a tube or sleeve 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 Cycle 11, implementation of the tube support plate interim plugging criteria limit requires a 100 percent 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 3.6 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:

SURVEILLANCE REQUIREMENTS (Continued)

| Category | Inspection Results |
|---|---|
| C-1 | Less than 5 percent of the total tubes inspected are degraded tubes and none of the inspected tubes are defective. |
| C-2 | One or more tubes, but not more than 1 percent of the total tubes inspected are defective, or between 5 percent and 10 percent of the total tubes inspected are degraded tubes. |
| C-3 | More than 10 percent of the total tubes inspected are degraded tubes or more than 1 percent of the inspected tubes are defective. |

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

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 following service under AVT conditions, 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 inservice inspection of a steam generator conducted in accordance with Table 4.4-2 requires a third sample inspection whose results fall in Category C-3, the inspection frequency shall be reduced to at least once per

SURVEILLANCE REQUIREMENTS (Continued)

20 months. The reduction in inspection frequency shall apply until a subsequent inspection demonstrates that a third sample inspection is not required.

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

2. A seismic occurrence greater than the Operating Basis Earthquake,

3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

4. A 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 percent of the nominal tube 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 imperfections greater than or equal to 20 percent of the nominal wall thickness caused by degradation.

4. Percent 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 containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

SURVEILLANCE REQUIREMENTS (Continued)

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 because it may become unserviceable prior to the next inspection. The plugging or repair limit imperfection depths are specified in percentage of nominal wall thickness as follows:

   a. Original tube wall 40%

   b. Babcock & Wilcox kinetic welded sleeve wall 40%

   c. Westinghouse laser welded sleeve wall 25%

   For Cycle 11, 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.10 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 4.4.5.3.c, 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 to the cold leg.

9. Tube Repair refers to sleeving which is used to maintain a tube in-service or return a tube to service. This includes the removal of plugs that were installed as a corrective or preventive measure. The following sleeve designs have been found acceptable:

   a. Babcock & Wilcox kinetic welded sleeves, BAW-2094P, Revision 1 including kinetic sleeve "tooling" and installation process parameter changes.

   b. Westinghouse laser welded sleeves, WCAP-13483, Revision 1.

SURVEILLANCE REQUIREMENTS (Continued)

10. Tube Support Plate Interim Plugging Criteria Limit is used for the disposition of a steam generator tube for continued service that is experiencing ODSCC confined within the thickness of the tube support plates. For application of the tube support plate interim plugging criteria limit, the tube's disposition for continued service will be based upon standard bobbin coil probe signal amplitude of flaw-like indications. Pending incorporation of the voltage verification requirements in ASME standard verifications, an ASME standard calibrated against the laboratory standard will be utilized in steam generator inspections for consistent voltage normalization. The plant specific guidelines used for all inspections shall be consistent with the eddy current guidelines in Appendix A of WCAP-14122.

    a. 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 c below is satisfied.

    b. 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 3.6 volts provided an RPC inspection does not detect degradation and item c below is satisfied.

    c. The projected distribution of crack indications is verified to result in total primary-to-secondary leakage less than 6.6 gpm in the most limiting loop during a postulated main steam line break event. The methodology for calculating expected leak rates from the projected crack distribution will be consistent with the latest EPRI recommended voltage-leak rate correlation described in WCAP-14122, using a probability of detection (POD) of 0.6.

    d. A tube with a flaw-like bobbin coil signal amplitude of greater than 3.6 volts shall be plugged or repaired.

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

SURVEILLANCE REQUIREMENTS (Continued)

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 and sleeve 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 and sleeves inspected.

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

3. Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

d. For Cycle 11, the results of inspection for all tubes in which the tube support plate interim plugging criteria has been applied shall be reported to the Commission pursuant to Specification 6.9.2 within 15 days following completion of the steam generator tube inservice inspection. The report shall include:

1. Listing of the applicable tubes, and

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

e. Projected Steam Line Break (SLB) Leakage performed under 4.4.5.4.a.10 will be reported to the Commission prior to restart of Cycle 11 (Mode 1).

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

| Preservice Inspection | No | | | Yes | | |
|---|---|---|---|---|---|---|
| No. of Steam Generators per Unit | Two | Three | Four | Two | Three | Four |
| First Inservice Inspection | All | | | One | Two | Two |
| Second & Subsequent Inservice Inspections | One (1) | | | One (1) | One (2) | One (3) |

Table Notation:

(1) The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N percent 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.

(2) The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in (1) above.

(3) 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 in (1) 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.<br><br>Notification to NRC pursuant to Specification 6.6 | All other S.G.s are C-1 | None | N/A | N/A |
| | | | Some S.G.s C-2 but no additional S.G.s 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 Specification 6.6. | N/A | N/A |

$S = \frac{3N}{n}\%$ 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 operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 1 gpm unidentified LEAKAGE,

c. 450 gallons per day total primary to secondary LEAKAGE through all steam generators,

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator, and

e. 10 gpm identified LEAKAGE.

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 next 30 hours.

b. With any Reactor Coolant System LEAKAGE greater than any one of the above limits, excluding pressure boundary LEAKAGE, 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.

### SURVEILLANCE REQUIREMENTS

4.4.6.2 Reactor Coolant System LEAKAGES shall be demonstrated to be within each of the above limits by:

a. Monitoring the following leakage detection instrumentation at least once per 12 hours:[1]

1. Containment atmosphere gaseous radioactivity monitor.

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(1) Only on leakage detection instrumentation required by LCO 3.4.6.1.

3/4.4.5 STEAM GENERATORS (Continued)

operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary LEAKAGE = 150 gallons per day per steam generator). Axial cracks having a primary-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 primary to secondary LEAKAGE of 150 gallons per day per steam generator can readily be detected. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or repair will be required of all tubes with imperfections exceeding the plugging or repair limit. Degraded steam generator tubes may be repaired by the installation of sleeves which span the degraded tube section. A steam generator tube with a sleeve installed meets the structural requirements of tubes which are not degraded, therefore, the sleeve is considered a part of the tube. The surveillance requirements identify those sleeving methodologies approved for use. If an installed sleeve is found to have through wall penetration greater than or equal to the plugging limit, the tube must be plugged. The plugging limit for the sleeve is derived from R.G. 1.121 analysis which utilizes a 20 percent allowance for eddy current uncertainty in determining the depth of tube wall penetration and additional degradation growth. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20 percent of the original tube wall thickness.

For Cycle 11, tubes experiencing outer diameter stress corrosion cracking at the tube support plates (TSPs) where such cracking is confined to the thickness of the TSPs will be dispositioned in accordance with Specification 4.4.5.4.a.10.

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 6.6 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.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

LCO (Continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

---------- NOTE 1 ----------

Maintaining an operating LEAKAGE limit of 150 gpd per steam generator (450 gpd total) for Cycle 11 will minimize the potential for a large LEAKAGE event during a main steam line break. Based on the non-destructive examination uncertainties, bobbin coil voltage distribution, and crack growth rate from the previous inspection, the expected leak rate following a steam line rupture is limited to below 6.6 gpm in the faulted loop. Maintaining LEAKAGE within the 6.6 gpm limit will ensure that offsite doses will remain within the 10 CFR 100 guidelines. LEAKAGE in the intact loops will be limited to the operating limit of 150 gpd. If the projected end-of-cycle distribution of crack indications results in primary-to-secondary LEAKAGE greater than 6.6 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service or repaired in order to reduce the postulated steam line break LEAKAGE to below 6.6 gpm.

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(1) c. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 450 gallons per day through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

(1) d. Primary to Secondary LEAKAGE through Any One SG

The 150 gallons per day limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

e. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of identified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.6.3, "RCS Pressure Isolation Valve (PIV)," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

a. If any pressure boundary LEAKAGE exists, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

ACTIONS (Continued)

b. Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB. If the unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS (SR)

SR 4.4.6.2

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

The RCS water inventory balance must be met with the reactor at steady state operating conditions and near operating pressure. Therefore, this SR is not required to be performed in MODES 3 and 4 until 12 hours of steady state operation near operating pressure have been established.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

SURVEILLANCE REQUIREMENTS (SR) (Continued)

SR 4.4.6.2 (Continued)

Steady state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be met when steady state is established. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the systems that monitor the containment atmosphere radioactivity and the containment sump level. The 12 hour monitoring of the leakage detection system is sufficient to provide an early warning of increased RCS LEAKAGE. These leakage detection systems are specified in LCO 3.4.6.1, "Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. Note (1) states that the 12 hour surveillance is required only on leakage detection instrumentation required by LCO 3.4.6.1. This Note allows the 12 hour monitoring to be suspended on leakage detection instrumentation which is inoperable or not required to be operable per LCO 3.4.6.1. Note (2) states that this SR is required to be performed during steady state operation.

3/4.4.6.3 PRESSURE ISOLATION VALVE LEAKAGE

The leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two in-series 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, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is identified LEAKAGE and will be considered as a portion of the allowed limit.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 184 TO FACILITY OPERATING LICENSE NO. DPR-66

DUQUESNE LIGHT COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 1

DOCKET NO. 50-334

## 1.0 INTRODUCTION

By letter dated July 29, 1994, as supplemented by a letter dated December 13, 1994, Duquesne Light Company, the licensee or DLC, submitted a request to change the Technical Specifications (TSs) for Beaver Valley Power Station, Unit 1. The proposed amendment revises, in part, TSs 4.4.5.2, 4.4.5.4, 4.4.5.5, 3.4.6.2 and Bases 3/4.4.5 and 3/4.4.6.2 to permit the use of voltage-based steam generator tube repair criteria for defects confined to within the thickness of the tube support plate. The proposed amendment also reduces the allowed primary-to-secondary operational leakage from any one steam generator from 500 gallons per day (gpd) to 150 gpd, and the total allowed primary-to-secondary leakage from all three steam generators from one gallon per minute (gpm), which is 1440 gpd, to 450 gpd. Whereas the proposed changes to the tube repair criteria are applicable to the eleventh operating cycle only, the proposed changes to the primary-to-secondary operational leakage limits are not restricted to one operating cycle.

The proposed voltage criteria pertain specifically to outside diameter stress corrosion cracking (ODSCC) flaws, and the proposed criteria: (1) permits flaws confined to within the thickness of the tube support plate with bobbin voltages less than or equal to 1.0 volt to remain in service; (2) permits flaws confined to within the thickness of the tube support plate with bobbin voltages greater than 1.0 volt but less than or equal to 3.6 volts to remain in service if a rotating pancake coil (RPC) probe does not detect degradation; and (3) requires flaw indications confined to within the thickness of the tube support plate with bobbin voltages greater than 3.6 volts to be plugged or repaired.

In their submittal, DLC noted that it had calculated the allowable primary to secondary leakage during a postulated main steam line break (MSLB) event, based on a small fraction of 10 CFR Part 100 limits at the site boundary. DLC further stated the projected steam line break leakage rate calculation methodology prescribed in draft NUREG-1477 was used.

Additional clarifying information with respect to implementation of the voltage-based repair criteria was provided in DLC's letter dated December 13, 1994. That information did not change the proposed TSs from the July 29,1994 letter, and therefore did not require a revision to the no significant

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hazards evaluation which was published in the Federal Register (59 FR 42337) on August 17, 1994. The NRC staff also conducted a conference call with the licensee on December 21, 1994, to discuss the issue of removing a sample of tubes from the steam generator for destructive examination.

1.1 Status of Staff Review Activities

The NRC staff is currently developing a generic interim position on voltage-based limits for ODSCC confined to within the thickness of the tube support plates. The staff has published several conclusions regarding voltage-based repair criteria in draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes" and in a draft generic letter (GL) titled "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes." The latter document was published for public comment in the Federal Register on August 12, 1994. However, the staff is continuing to evaluate an acceptable generic position which will take into consideration public comments on the draft GL cited above, domestic operating experience under the voltage-based repair criteria, and additional data which have been made available from European nuclear power plants. The staff currently plans to document its final position on this matter in a GL. Pending completion and issuance of the staff's final generic position on the voltage-based tube repair criteria, the staff is continuing to evaluate voltage-based repair criteria proposals on a plant-specific basis, as necessary, to ensure that there is adequate assurance of public health and safety. Furthermore, these plant-specific evaluations limit the applicability of the voltage-based repair criteria to one cycle of operation.

The tube repair limits proposed by DLC include a lower voltage repair limit of 1.0 volt for axially oriented ODSCC flaws confined to within the thickness of the tube support plates in lieu of the present criterion which is a depth-based limit of 40%. In addition, the repair limits allow bobbin indications between 1.0 and 3.6 volts (the upper voltage repair limit) to remain in service provided inspection of these indications with a rotating pancake coil (RPC) probe does not confirm the degradation to be present.

DLC's proposal is similar to that reviewed and approved for several other plants and has been reviewed on a plant-specific basis. The tube structural limit is based on maintaining a margin of safety of 1.43 against tube failure under postulated accident conditions and maintaining a margin of safety of 3 against burst during normal operation. The margin of safety of 3 against burst during normal operation is inherently satisfied since the structural constraint provided by the tube support plates, which surround the degradation to which the voltage-based repair criteria applies during normal operation, ensures these tubes will maintain this margin of safety at these locations. To complement these deterministic criteria, the conditional probability of burst under accident conditions and the primary-to-secondary leakage from the steam generator tubes during a postulated main steam line break (MSLB) are also calculated.

## 1.2 Proposed Interim Tube Repair Criteria

Beaver Valley, Unit 1, TSs 4.4.5.2, 4.4.5.4, 4.4.5.5, 3.4.6.2 and Bases 3/4.4.5 and 3/4.4.6.2, are revised by this amendment request to specify the tube repair and leakage criteria for ODSCC confined to within the thickness of the tube support plate. The proposed changes to the tube repair and leakage criteria in the TSs specify, in part:

a. Implementation of the tube support plate plugging criteria 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 ODSCC indications.

b. 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 (d) below is satisfied.

c. 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 3.6 volts provided an RPC inspection does not detect degradation and item (d) below is satisfied.

d. The projected distribution of crack indications is verified to result in total primary-to-secondary leakage less than 6.6 gpm in the most limiting loop during a postulated main steam line break (MLSB) event.

e. A tube with a flaw-like bobbin coil signal amplitude of greater than 3.6 volts shall be plugged or repaired.

f. Projected MSLB leakage will be reported to the Commission prior to restart of Cycle 11.

g. Reactor coolant system leakage shall be limited, in part, to 450 gpd total primary-to-secondary leakage through all steam generators and 150 gpd through any one steam generator.

In addition to the above TS changes, DLC also made the following commitments for implementing the voltage-based repair criteria:

1. All bobbin indications with voltages greater than 1.0 volt will be inspected using an RPC probe to confirm axial ODSCC as the dominant mechanism for indications at the tube support plates.

2. Tubes with bobbin dent voltages exceeding 5.0 volts will be inspected with an RPC and any RPC flaw indications detected at these intersections will be repaired.

3. An RPC sample inspection of tubes with artifact/large mixed residual signals that could potentially mask a 1.0 volt bobbin signal will be performed. Any RPC flaw indications detected at these intersections will be plugged or repaired.

4. Tubes with known leaks will be repaired prior to returning the steam generators to service.

5. Steam generator tube integrity data (i.e., voltage distributions and leak/burst evaluations) will be provided to the NRC within 90 days following restart.

6. A 0.720-inch diameter bobbin coil probe will be used during the steam generator inspections at intersections where the voltage-based repair criteria will be applied. If an alternate probe size is found to be required, appropriate performance demonstration test results will be provided.

7. The NRC will be notified prior to plant restart if any primary water stress corrosion cracking (PWSCC) indications are detected at the tube support plate elevations during the steam generator inspections.

8. The conditional probability of burst calculation will consider parametric uncertainty.

9. The NRC will be informed, prior to plant restart from the refueling outage, of any unexpected inspection findings relative to the assumed characteristics of the flaws at the tube support plate intersections. This includes any detectable circumferential indications or detectable OD indications extending outside the thickness of the tube support plate.

10. Six tube support plate intersections will be removed during the outage scheduled to begin January 2, 1995, as required by the NRC, for destructive analysis.

## 2.0 EVALUATION

### 2.1 Inspection Issues

In support of the proposed voltage-based repair limits, DLC proposes to utilize the eddy current test guidelines provided in its license amendment proposal. The inspection criteria are intended to ensure the inspection scope, data acquisition, and data analysis are performed in a manner consistent with the methodology utilized to develop the voltage limits. The proposed guidelines define, in part, the bobbin specifications, calibration requirements, specific acquisition and analyses criteria, and flaw recording guidelines to be used for the inspection of the steam generators.

The inspections to be performed as part of the voltage-based repair criteria include both bobbin coil and RPC examinations. Bobbin coil examinations will be performed for 100% of the hot-leg tube support plate intersections and cold-leg intersections down to the lowest cold-leg tube support plate with known ODSCC. Furthermore, DLC plans to inspect all active tubes in each steam generator from tube-end to tube-end. The bobbin coil examinations will be performed with a 0.720-inch bobbin coil probe pending a performance demonstration that smaller/larger diameter bobbin coil probes provide, on a statistically significant basis, an equivalent voltage response and detection capability when compared to the primary size probe (i.e., 0.720-inch probe). RPC examinations will be performed to permit additional characterization of the flaws found with the bobbin coil probe and to inspect intersections with significant bobbin interference signals (due to copper deposits, dents, large mix residuals) which may impair the ability of the bobbin coil probe to detect flaws or which may unduly influence the bobbin voltage measurement.

With respect to flaw characterization, a key purpose of the RPC inspections is to ensure the absence of detectable crack-like circumferential indications and detectable indications extending outside the thickness of the tube support plate. The voltage-based repair criteria are not applicable to intersections exhibiting such indications (i.e., circumferential indications and indications extending outside the tube support plates), and special reporting requirements pertaining to the finding of such indications have been proposed if these types of indications are detected. RPC examinations will be performed: (1) at all intersections with bobbin coil indications exceeding 1.0 volt; (2) at all intersections where the dent signal is greater than 5.0 volts; and (3) at intersections where the mixed residual could cause a 1.0 volt bobbin signal to be missed or misread (i.e., masked). Any flaw-like indications found at intersections with dent signals greater than 5.0 volts or large mixed residuals will result in tube repair. Since DLC does not currently have copper in the secondary system and has not previously observed significant copper interference in the eddy current signal at the tube support plate elevation, copper deposits are not expected to be observed and hence RPC sampling of such locations is not applicable.

As previously mentioned, tube support plate locations with bobbin dent voltages above 5.0 volts will be inspected with an RPC probe. Inspections of dented intersections are performed, in part, as a result of: (1) the possible masking effect the dent may have on the detection of flaw indications; (2) the possible development of primary water stress corrosion cracking (PWSCC) flaws at these locations; and (3) the possible development of circumferential cracks at these locations. With respect to masking flaw indications, it is anticipated that flaw signals on the order of 1.0 volt would have phase angles that fall within the flaw reporting range even if the bobbin dent voltage was as high as 5.0 volts based on a vectorial combination of the eddy current signals attributed to the flaw and to the dent. As a result, RPC inspecting all intersections with bobbin dent voltages in excess of 5.0 volts provides reasonable assurance that any structurally significant ODSCC indications will

be detected and repaired. With respect to the occurrence of circumferential cracking at the support plate elevations, the RPC sampling plan provides assurance that if a significant amount of circumferential cracking is occurring at the tube support plate elevations, it will be detected.

With respect to the occurrence of PWSCC at dented tube support plate intersections, the potential exists for axial PWSCC to occur at intersections where the bobbin dent voltage is less than 5.0 volts. Most frequently these types of indications (i.e., indications representative of axially oriented PWSCC) have been found at tube support plates with significant denting, have been known to occur at 180° spacing as two axial indications due to the stresses in the tube, and have been known to occur within the tube support plate but occasionally extending outside the tube support plate. Axial PWSCC is not presently analyzed as part of the voltage-based repair criteria. As a result of this and the potential for PWSCC to occur at dented intersections less than 5.0 volts, the licensee has proposed to: (1) RPC inspect all bobbin indications which are greater than 1.0 volt at dented intersections; (2) RPC inspect all intersections where the bobbin dent voltage is greater than 5.0 volts regardless of whether a bobbin indication is detected; and (3) notify the NRC prior to plant restart if any PWSCC indications are detected at the support plate elevations. In addition, DLC will brief the eddy current analysts on the potential for PWSCC at tube support plate locations and the analysts will be instructed to report occurrences of axial PWSCC. The staff finds this sampling plan adequate to detect the onset of axial PWSCC at support plate locations. The staff also notes that frequently axial PWSCC extends outside the tube support plate intersection, making it more likely to be detectable with the bobbin coil. This provides added confidence that if extensive axial PWSCC is present, it will be detected. The staff notes that if PWSCC is detected at support plate elevations, an evaluation to ensure the voltage-based repair criteria is only applied to ODSCC indications will need to be performed and reviewed by the staff.

With respect to data acquisition and analysis, DLC's eddy current guidelines either contain requirements or guidance pertaining to: (1) recording all indications regardless of voltage amplitude; (2) controlling probe wear by the use of a probe wear standard; (3) calibrating the bobbin coil probes; and (4) using a transfer standard to ensure consistency between the voltages measured in the field and the voltages measured in the laboratory as part of the development of the voltage-based approach.

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 and in the draft GL cited above) which will be resolved prior to issuing the final GL on voltage-based limits for ODSCC confined to within the thickness of the tube support plate. These outstanding issues include, in part: (1) limits on new probe variability; (2) the need to reinspect all tubes since the last successful probe wear check; (3) the need to calibrate the bobbin coil on the 4-100% holes versus the 4-20% holes; and (4) the capabilities/limitations of the 1-coil, 2-coil, and 3-coil RPC probes. However, the staff concludes that the inspection guidelines

submitted by DLC are acceptable since the proposed repair criteria are limited to one cycle, and the calibration, recording, and analysis requirements are consistent with the methodology used in the development of the databases and supporting evaluations.

## 2.2 Tube Integrity Issues

The thin-walled tubing of the steam generator constitutes more than half of the reactor coolant pressure boundary (RCPB), and maintenance of the structural and leakage integrity of this boundary is a requirement under Title 10 of the Code of Federal Regulations Part 50 (10 CFR Part 50), Appendix A. Specific requirements governing the maintenance of steam generator tube integrity are contained in the plant TSs and Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). These include requirements for periodic inservice inspection of the tubing, flaw acceptance criteria (i.e., repair limits for plugging or sleeving), and primary-to-secondary leakage limits. These requirements, coupled with the broad scope of plant operational and maintenance programs, have formed the basis for assuring adequate steam generator tube integrity.

Flaw acceptance criteria, termed plugging/repair limits, are specified in the plant TSs. The purpose of the TS repair limits 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, 30, 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 the objectives of the General Design Criteria related to steam generator tube integrity has been to establish a minimum wall thickness requirement in accordance with the structural criteria of Regulatory Guide (RG) 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes." Allowances for eddy current measurement error and flaw growth between inspections have been added to the minimum wall thickness requirements, consistent with RG 1.121, to arrive at a depth-based repair limit. Development of the minimum wall thickness requirements to satisfy RG 1.121 was governed by analyses for uniform thinning of the tube wall in the axial and circumferential directions. The assumption of uniform thinning conservatively bounds the degrading effects of all flaw types currently occurring in the field and is the basis of the standard 40% depth-based repair limit incorporated into the TSs. However, the 40% repair limit is conservative for highly localized flaws such as pits and short cracks. In particular, the 40% depth-based repair limit is conservative for ODSCC that occurs at the tube support plate intersections.

Enforcement of a minimum wall thickness requirement for the steam generator tubes would implicitly serve to ensure leakage integrity during normal operation and postulated accidents, as well as structural integrity. It has been recognized, however, that defects, especially cracks, may occasionally

grow entirely through-wall and develop small leaks. For this reason, limits on the allowable primary-to-secondary leakage have been established in a plant's TSs to ensure timely plant shutdown before adequate structural and leakage integrity of an affected steam generator tube is impaired.

The proposed voltage-based tube repair limits consist of voltage amplitude criteria rather than the traditional depth-based criteria. Thus, the repair criteria represent a departure from the past practice of explicitly enforcing a minimum wall thickness requirement.

The industry-wide database from examination of steam generator tubes removed from a number of steam generators in operating nuclear power plants shows that for bobbin indications exceeding 1.0 volt (i.e., the lower voltage repair limit), maximum crack depths range between 20% and 98% 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 3.6 volts (i.e., the upper voltage repair limit), the maximum crack depths have been found to generally range between 60% and 100% through-wall. Many of the tubes which will be allowed to remain in service under the proposed voltage-based repair criteria 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 2.3 and 2.4 of this evaluation.

Although the voltage-based repair limits ensure adequate structural and leakage integrity, the NRC staff recognizes that overall margins have been reduced when compared to the margins associated with the existing 40% depth-based repair limit. Because of the increased likelihood of through-wall cracks developing in service, the staff has included provisions for augmented steam generator inspections, as discussed in the previous section, and more restrictive operational tube leakage limits, as discussed below.

## 2.3 Structural Integrity

### 2.3.1 Deterministic Structural Integrity Assessment

DLC has proposed a burst pressure/bobbin voltage correlation to demonstrate that bobbin indications satisfying the 1.0 volt lower voltage repair limit would retain adequate structural margins, consistent with the criteria of RG 1.121. The correlation was developed from both pulled steam generator tube data from other 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 and are consistent with the calibration standard voltage set-ups and voltage measurement procedures to be used by DLC during the steam generator inspections.

To confirm the nature of the degradation occurring at the tube support plate elevations, DLC pulled two tubes with six tube support plate intersections from the steam generators during an outage in 1991. Tube pulls confirm that the nature of the degradation being observed at the tube support plate elevations is predominantly axially oriented ODSCC and also provide data for assessing the reliability of the inspection methods and for supplementing existing databases (e.g., burst pressure, probability of leakage, and leak rate). Metallurgical examinations were performed on five of the six tube support plate intersections removed from the steam generators. The sixth intersection was archived for any future analysis which may be needed. The examinations performed confirmed that in 1991 the dominant degradation mechanism for the indications at the support plate elevations was axially oriented ODSCC and that the indications at these intersections are consistent with the data in the databases accumulated from other plants with ODSCC at tube support plate intersections.

The draft GL on voltage-based repair criteria provides guidance on performing tube pulls for initial implementation of the repair criteria. In summary, the draft GL states that at least six tube support plate intersections should be obtained either during the outage in which the voltage-based repair criteria are implemented or during the inspection outage preceding initial application of the voltage-based repair criteria. As a result, the staff has concluded that implementation of the voltage-based repair criteria at Beaver Valley Unit 1 is contingent upon DLC removing six tube support plate intersections for destructive analysis during this outage. The issue of tube pulls was discussed with DLC during a conference call on December 21, 1994. DLC has now completed these tube pulls with the required 6 tube support plate intersections.

The voltage repair criteria previously approved by the staff 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 projected to occur during the next operating cycle. Because the voltage-based repair criteria addresses tubes affected with ODSCC confined to 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 to which the voltage-based criteria apply will retain a margin of 3 with respect to burst under normal operating conditions, consistent with the criteria of RG 1.121. For a postulated MSLB accident, however, the tube support plates may displace axially during blowdown such that the ODSCC affected portion of the tubing may no longer be fully constrained by the tube support plates. Accordingly, it is appropriate to consider the ODSCC affected regions of the tubes as free standing tubes for the purpose of assessing burst integrity under postulated MSLB conditions.

The allowable end-of-cycle (EOC) voltage which ensures a margin of 1.43 with respect to burst under postulated MSLB conditions (i.e., 3660 psi), in accordance with RG 1.121, is based on the lower 95% prediction interval of the burst pressure/bobbin voltage correlation, adjusted for lower bound material

properties evaluated at the 95/95 confidence level. This voltage limit is approximately 8.8 volts for the 7/8-inch diameter tubing used in the Beaver Valley 1 steam generators. The difference between the 8.8 volt allowable EOC voltage and the 1.0 volt repair criterion represents an allowance of 7.8 volts for voltage growth (i.e., ODSCC flaw growth) during the forthcoming fuel cycle (Cycle 11) and for eddy current voltage measurement variability (i.e., the repeatability error) during the steam generator inspection.

To demonstrate the adequacy of the voltage-based repair criterion, the largest RPC confirmed indication which may be left in service (i.e., a 1.0 volt indication), was analyzed by the staff to determine if the indication would grow to the point that the structural voltage limit (i.e., 8.8 volts) is exceeded. In this analysis, a 1.0 volt bobbin indication is assumed to grow at a rate equal to the maximum growth rate observed during the latest cycle for which data are available (i.e., 1.2 volts for Cycle 9 which was 1.35 EFPY in duration) and it is assumed that the indication was undersized by 20% (i.e., the 95% cumulative probability of the non-destructive examination (NDE) uncertainty). The resultant EOC voltage is determined from this analysis to be 2.0 volts for the 0.95 EFPY planned for Cycle 11. This EOC voltage compares favorably to the structural voltage limit of 8.8 volts determined from the burst pressure versus bobbin voltage correlation.

The proposed 1.0 volt lower repair limit is applicable to all bobbin indications confirmed by RPC or which have not been RPC inspected. DLC is also proposing a 3.6 volt upper voltage repair limit applicable to bobbin indications which have been RPC inspected but for which the RPC failed to confirm the bobbin indication. This 3.6 volt upper voltage repair limit was derived in WCAP-12871 Revision 2, "J.M. Farley Units 1 and 2, SG Tube Plugging Criteria for ODSCC at Tube Support Plates." In WCAP-12871 Revision 2, the upper voltage repair limit was based on a structural limit of three times the normal operating pressure differential. The maximum voltage which would satisfy this burst pressure criterion for tubes with lower bound material properties at a 95% prediction interval was 6.2 volts based on the data available at that time. A 3.6 volt upper voltage repair limit was calculated from the 6.2 volt structural limit by including an allowance for average growth rates of 50% of the BOC voltage amplitude and an allowance of 20% for eddy current voltage measurement variability (i.e., the 95% cumulative probability of the NDE uncertainty).

Since the issuance of WCAP-12871 Revision 2 in February 1992, additional data have been added to the burst pressure database used in the development of this upper voltage repair limit and several of the existing data points in the database have been updated as a result of additional analysis. In addition, it has been determined that the voltage limit should be derived from a structural limit of 1.43xMSLB pressure since the constraint provided by the tube support plate during normal operation results in the three times the normal operating differential pressure criteria of RG 1.121 being inherently satisfied for this degradation mechanism. This has resulted in a new upper voltage repair limit of 5.5 volts for Beaver Valley Unit 1 assuming an allowance of 40% for flaw/voltage growth over the next operating cycle (Cycle 11) and an allowance of 20% for measurement variability. The

voltage measurement variability estimate considers measurement variabilities stemming from bobbin coil probe wear and variability in the analysts' interpretation of the bobbin coil voltage. Potential flaw growth between inspections has been evaluated based on observed voltage amplitude changes during prior cycles at Beaver Valley Unit 1. Over the last few cycles the average percent voltage growth at Beaver Valley Unit 1 has been 44% (1987 to 1989), 19% (1989 to 1991), and 16% (1991 to 1993). The 40% average growth allowance used to support the 5.5 volt upper voltage repair limit is intended to provide margins for variation in future growth rates at Beaver Valley Unit 1. As a result of the above analysis, the staff concludes that the 3.6 volt upper voltage repair limit is acceptable for Beaver Valley Unit 1.

The staff has evaluated the acceptability of the upper voltage repair limit for indications below this limit which may be left in service if detected by the bobbin coil probe but not confirmed to be flaw-like by the RPC probe. Short and/or relatively shallow cracks detected by the bobbin coil may sometimes not be detectable by the RPC probe, although the RPC probe is considered by the staff to be more sensitive to longer, deeper flaws which are of structural significance. Furthermore, the burst strength of steam generator tubing affected by predominantly axially oriented ODSCC at the support plate elevations is not a unique function of the bobbin voltage. Rather, for a given voltage, there is a statistical distribution of possible burst strengths, as indicated in the burst pressure/bobbin voltage correlation. The staff believes that the burst pressure for bobbin indications which were not confirmed to be flaw-like by the RPC probe will tend to be at the upper end of the burst pressure distribution (i.e., exhibit a higher burst pressure). That is, ODSCC which is not detectable by RPC is believed to be less likely to affect the tube structural and leakage integrity during the operating cycle than ODSCC which is detectable by both the bobbin coil and the RPC probe. In addition, the burst and leakage potential for bobbin indications accepted for continued service under the 3.6 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 considerations, the staff finds the upper voltage repair limit of 3.6 volts for indications which may be left in service if detected by bobbin inspection but not confirmed by the RPC to be acceptable.

### 2.3.2 Probabilistic Structural Integrity Assessment

A probabilistic analysis for the potential for steam generator tube ruptures, given a MSLB, must also be performed. The need for this analysis, which supplements the deterministic analysis discussed above, is dictated by the following considerations:

1. The deterministic analysis does not consider the tail of the burst pressure distribution beyond the lower 95% prediction interval used to determine the maximum allowable EOC voltage. Given the large numbers of

indications being accepted for continued service with the 1.0 volt criterion, the probabilistic analysis ensures that the use of the 95% prediction interval value in lieu of the 99% or 99.9% values does not lead to a significant likelihood of steam generator tube rupture given a MSLB.

2. The deterministic assessment ignores the burst and leakage potential of bobbin indications between 1.0 volt and 3.6 volts for which the RPC probe failed to confirm the indication. The probabilistic assessment, however, considers the burst potential of these indications with no credit given for the lack of confirmation by the RPC probe of the presence of these indications.

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

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

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

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

below the MSLB differential pressure for the EOC voltage distribution by the total number of samples. A distribution of material tensile properties is sampled in the probabilistic analysis since the data points in the bobbin voltage/burst pressure correlation have been normalized to a flow stress of 75 ksi.

The POD scaling approach cited above is reasonably consistent with reported operating experience to-date with ODSCC in terms of accounting for the projected distribution of indications at EOC which were not previously detectable at BOC. However, operating experience to-date, for ODSCC confined to within the thickness of the tube support plate, is that maximum EOC bobbin voltages generally do not exceed 4 or 5 volts. Although there are known cases where indications on the order of 3 volts have not been detected, there is very little experience regarding the likelihood of not detecting bobbin indications between 3 and 10 volts. The industry believes that the numerical value of the POD is substantially higher than 0.6 for indications exceeding 1.0 volt, based, in part, on data collected from the Electric Power Research Institute (EPRI) performance demonstration program. However, pending further staff review, the staff believes a POD value of 0.6 is appropriate for this voltage-based repair criteria application.

DLC will perform the probabilistic analysis discussed above which assumes the degradation is free span and ignores the potential constraining effects of the tube support plates. In addition, this analysis will be performed in a manner which considers the uncertainty in the parameters for the supporting correlations (e.g., burst pressure/bobbin voltage correlation). The results of the probabilistic analysis will be compared to a threshold value established by the staff. Consistent with the draft GL this threshold value is $1x10^{-2}$. This threshold value will provide assurance that the probability of burst is acceptable considering the assumptions of the calculation and the results of the staff's generic risk assessment for steam generators contained in NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." Failure to meet the threshold value indicates that ODSCC confined to within the thickness of the tube support plate could contribute a significant fraction to the overall conditional probability of tube rupture from all forms of degradation that was assumed and evaluated as acceptable in NUREG-0844. In addition, the threshold value provides an indication that one or more tubes may not maintain the RG 1.121 safety margins for the entire operating cycle. In a submittal dated December 13, 1994, DLC stated that if requested by the NRC, the results of this calculation will be reported to the staff prior to restart from the refueling outage. The staff hereby requests that the results of this calculation be provided to the staff prior to plant restart. If the results of the probability of burst analysis exceed the $1x10^{-2}$ threshold value, the staff will request the licensee to assess the safety significance of this occurrence prior to plant restart.

### 2.3.3 Data Exclusion from the Burst Pressure Correlation

During the performance of the pulled tube examinations, malfunctions in the test equipment or improper specimen preparation can occasionally occur which could result in erroneous readings. Data such as this should not be included in a database since it could result in invalid results and/or conclusions. The staff, therefore, concluded in draft NUREG-1477 that eliminating data from the bobbin voltage/burst pressure database was appropriate provided that the data could be shown to be erroneous or the result of an invalid test. The staff provided additional guidance regarding the exclusion of data from the correlations used in the bobbin voltage/burst pressure database in a meeting with the industry on February 8, 1994. As a result of this guidance, the industry provided criteria for determining whether data may be removed from the burst pressure/bobbin voltage database. The specific criteria are presented in a letter referenced by DLC which was submitted to the NRC by EPRI on April 22, 1994.

The data points excluded from the burst pressure/bobbin voltage database as a result of applying these criteria are listed in Table E-1 of the subject document. The staff has concluded that excluding the data points listed in Table E-1 from the 7/8-inch diameter steam generator tubing burst pressure/bobbin voltage database is appropriate. Pending further evaluation of the generic criteria presented in Section E.2, the staff is continuing to assess the appropriateness of excluding data points from the burst pressure correlation on a case-by-case basis.

### 2.3.4 Combined Accident Loadings

DLC has evaluated the effects of combined accident loads such as safe shutdown earthquake (SSE) plus loss-of-coolant accident (LOCA) loads and SSE plus main steam line break (MSLB) loads on tube integrity consistent with General Design Criteria (GDC) 2 of 10 CFR Part 50, Appendix A. These evaluations are provided in a Westinghouse proprietary report, WCAP-13579, "Beaver Valley Unit 1 Steam Generator Tube Plugging Criteria for Indications at Tube Support Plates." A combined LOCA plus SSE must be evaluated for potential yielding of the tube support plates which could result in subsequent deformation of the tubes. If significant tube deformation should occur, primary flow area could be reduced and postulated cracks in tubes could open up which might create the potential for in-leakage (i.e., secondary-to-primary leakage) under LOCA conditions. In-leakage during a LOCA would pose a potential concern since it may cause an increase in the core peak clad temperature (PCT).

The most limiting accident conditions for tube deformation considerations result from the combination of SSE and LOCA loads. The seismic excitation defined for steam generators is in the form of acceleration response spectra at the steam generator supports. In the seismic analysis, the licensee has used generic response spectra which envelop the Beaver Valley specific response spectra. A finite element model of the Series 51 steam generator was developed and the analysis was performed using the WECAN computer program. The mathematical model consisted of three dimensional lumped mass, beam, and pipe elements as well as general matrix input to represent the piping and

support stiffness. Interactions at the tube support plate shell and wrapper/shell connections were represented by concentric spring-gap dynamic elements. Impact damping was used to account for energy dissipation at these locations.

Based on prior qualification of the Beaver Valley Unit 1 primary piping for leak-before-break requirements, a break in the primary piping is not likely. DLC, however, has used the loads for the primary piping break as a conservative approximation. The principal tube loading during a LOCA is caused by the rarefaction wave in the primary fluid. This wave initiates at the postulated break location and travels around the tube U-bends. A differential pressure is created across the two legs of the tube which causes an in-plane horizontal motion of the U-bends and induces significant lateral loads on the tubes. The pressure time histories needed for creating the differential pressure across the tube are obtained from transient thermal-hydraulic analyses using the MULTIFLEX computer code. For the rarefaction wave-induced loadings, the predominant motion of the U-bends is in the plane of the U-bend. Thus, the individual tube motions are not coupled by the anti-vibration bars and the structural analysis is performed using single tube models limited to the U-bend and the straight leg region over the top two tube support plates.

In addition to the rarefaction wave loading discussed above, the tube bundle is subjected to bending loads during a LOCA. These loads are due to the shaking of the steam generator caused by the break hydraulics and reactor coolant loop motion. However, the resulting tube support plate loads from this motion are small compared to those due to the rarefaction wave-induced motion.

A dynamic blowdown analysis was performed to obtain the LOCA-induced hydraulic forcing function. A double-ended guillotine break (1.0 msec break opening time) was assumed in this analysis. The hydraulic forcing function was applied to a system structural model that includes the steam generator, the reactor coolant pump, and the primary piping. This analysis yields the time-history displacements of the steam generator and its upper lateral and lower support nodes. These time-history displacements formulate the forcing functions for obtaining the tube stresses due to LOCA shaking of the steam generator.

In calculating a combined tube support plate load, DLC combined the LOCA rarefaction and LOCA shaking loads directly, while the LOCA and SSE loads were combined using the square-root of the sum-of-the-squares. The staff found this combination methodology acceptable. The overall tube support plate load was transferred to the steam generator shell through wedge groups located at discrete locations around the plate circumference.

The radial loads due to combined LOCA and SSE could potentially result in yielding of the tube support plate at the wedge supports, causing some tubes in the vicinity of the wedge supports to be deformed. Utilizing results from recent tests and analysis programs, DLC has shown that tubes will undergo permanent deformation if the change in diameter exceeds a minimum threshold

value. This threshold for tube deformation is related to the concern for tubes with preexisting tight cracks that could potentially open during a combined LOCA plus SSE event.

The data provided by DLC for Beaver Valley Unit 1 indicates that LOCA plus SSE loads would be of such magnitude that none of the tubes (which are assumed to contain preexisting tight cracks) are predicted to exceed the deformation threshold value discussed above. Therefore, none of the deformed tubes will leak significantly.

DLC has assessed the effect of SSE bending stresses on the burst strength of tubes with axial cracks. Tensile stress in the tube wall would tend to close the cracks while compressive stress would tend to open the cracks. On the basis of previously-performed tests, DLC has concluded that the burst strength of tubes with through-wall cracking is not affected by an SSE event.

Based on a review of the information provided by DLC, the staff concludes that at Beaver Valley Unit 1, no significant tube deformation or leakage is likely to occur during an SSE plus LOCA event. In addition, the burst strength of tubing with through-wall cracks is not affected by an SSE event. Therefore, no tubes need to be excluded from IPC for reasons of deformation resulting from combined accident loads.

## 2.4 Leakage Integrity

An important implication of voltage-based steam generator tube repair criteria is that the criteria may permit tubes to have, or to develop, through-wall or near through-wall cracks during the forthcoming operational cycle, thus creating the potential for primary-to-secondary leakage during normal operation, transients, or postulated accidents. Thus, the leakage integrity of these tubes, in addition to their structural integrity, must be assessed.

The staff finds that adequate leakage integrity during normal operating conditions is reasonably assured by the TS limits on allowable primary-to-secondary leakage. Adequate leakage integrity during transients and postulated accidents is demonstrated by showing that for the most limiting accident, assumed to occur at the end of the next operating cycle, the resulting leakage will not exceed a rate that will result in offsite dose limits being exceeded. The radiological consequences of this is discussed in Section 2.5.

### 2.4.1 Normal Operational Leakage

Implementation of the voltage-based tube repair criteria includes a reduction in the TS reactor coolant system leakage limits. Specifically, the present TS limit of 500 gallons per day (gpd) for primary-to-secondary leakage through any one steam generator is reduced to 150 gpd, and the limit on the total primary-to-secondary leakage through all steam generators is reduced from 1.0 gpm, which is 1440 gpd, to 450 gpd.

The present 500 gpd limit per steam generator is intended to ensure that through-wall cracks which leak at rates up to this limit during normal operation will not propagate and result in tube rupture under postulated accident conditions consistent with the criteria of RG 1.121. Development of the 150 gpd per steam generator leakage limit has utilized the extensive industry database regarding burst pressure as a function of crack length and leakage during normal operation. Based on leakage evaluated at the lower 95% confidence interval for a given crack size, the 150 gpd limit would be exceeded before the crack length reaches the critical crack length for MSLB pressures. Based on nominal, best estimate leakage rates, the 150 gpd limit would be exceeded before the crack length reaches the critical crack length corresponding to a burst pressure of three times normal operating pressure.

The reduced steam generator leakage limits to be adopted for implementation of the voltage-based tube repair criteria are more restrictive than the present operating leakage limits in the plant's TSs in order to provide a margin of safety against rupture. This reduction in the steam generator maximum allowable leakage limits is also intended to provide an additional margin in the event that a crack grows at a rate much greater than expected or which may unexpectedly extend outside the thickness of the tube support plate. The staff finds the proposed operating leakage limits in TS 3.4.6.2 to be acceptable for implementation of the voltage-based tube repair criteria.

2.4.2 Accident Leakage

The licensee has proposed a model for calculating the steam generator tube leakage from the faulted steam generator during a postulated MSLB which consists of two major components: (1) a model predicting the probability that a given indication will leak as a function of voltage (i.e., the probability of leakage (POL) model); and (2) a model predicting leak rate as a function of voltage, given that leakage occurs (i.e., the conditional leak rate model).

In the POL model, the probability that a given indication will leak is presented as a function of the bobbin coil voltage of that indication. The data are separated into two categories (i.e., indications which leak during a MSLB and those which do not). While various functional forms can be fitted to the data, the staff has concluded that a single functional form, the log-logistic, is acceptable for the purpose of assessing MSLB-induced steam generator tube leakage. The staff believes that any nonconservatism associated with the use of the log-logistic model, as compared to the other functional forms, is small compared to the conservatism inherent in the existing methodology for calculating the steam generator tube leakage and the radiological consequences of this leakage induced by a postulated MSLB. In addition, the differences in the POL functional forms are considered to be less significant when the leakage is calculated using a linear leak rate model, as discussed below, instead of a constant leak rate model which treats leakage as independent of voltage.

Regarding the conditional leak rate model, a correlation between the steam generator tube leak rate and bobbin voltage data based on a linear regression fit of the logarithms of the data has been developed. The staff provided statistical criteria in the draft GL on voltage-based repair criteria which permits licensees to use such a correlation if the correlation can be statistically justified at a 95% confidence level (i.e., a p-value of 5%). DLC has proposed to use such a correlation for calculating the leakage during postulated accident conditions. The staff concludes that using a linear relationship between the logarithms of the leak rate and bobbin voltage is appropriate in the determination of the primary-to-secondary steam generator tube leakage during a postulated MSLB provided the statistical criteria delineated in the draft GL on this subject are met. If the statistical criteria in the draft GL are not met, the linear regression should be assumed to have zero slope (i.e., the linear regression fit should be assumed to be constant with voltage). The staff further notes that the databases used in such evaluations should be consistent with the databases discussed in Section 2.4.3 of this evaluation.

DLC has proposed a method for determining the primary-to-secondary steam generator tube leakage during a postulated MSLB which involves a Monte Carlo method which simulates the regression parameter uncertainties. The staff has analyzed this model for the case where the p-value test is valid at the 5% level, and has concluded that this model is appropriate and consistent with the draft GL on voltage-based repair criteria. This method involves:

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

2. Using the regression parameters from Step 1 to determine the leak rate for each flaw indication in the estimated EOC voltage distribution. The EOC voltage distribution used in this calculation is the same as that discussed in Section 2.3.2.

3. Calculating the sum of the individual leak rates determined in Step 2 to obtain a value of the total steam generator leak rate.

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

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

The staff notes that some minor variations in the details of the modeling may be necessary for the case where the p-value test is invalid at the 5% level. If the p-value test is invalid, these variations should be discussed in the licensee's submittal providing the steam generator tube integrity data as discussed in section 1.2.

DLC has calculated the allowable steam generator leak rate to be 6.6 gpm in the faulted steam generator. This value is consistent with maintaining the radiological consequences of a release outside containment to within a small fraction of the guideline values in 10 CFR Part 100. As a result, if the primary-to-secondary leakage during a postulated MSLB is less than the 6.6 gpm limit, the steam generator tubing will maintain adequate leakage integrity under these conditions.

2.4.3 Data Exclusion from the Leakage Correlations

During the performance of the pulled tube examinations, malfunctions in the test equipment or improper specimen preparation can occasionally occur which could result in erroneous readings. Data such as these should not be included in the database since they could result in invalid results and/or conclusions. The staff, therefore, concluded in draft NUREG-1477 that eliminating data from the conditional leak rate and probability of leakage databases was appropriate provided that the data could be shown to be erroneous or the result of an invalid test. The staff provided additional guidance regarding the exclusion of data from the databases used in the steam generator tube leakage evaluation in a meeting with the industry on February 8, 1994. As a result of this guidance, the industry provided criteria for determining whether data may be removed from the probability of leakage and conditional leak rate databases. The specific criteria are presented in a letter referenced by DLC which was submitted to the NRC by EPRI on April 22, 1994.

The data points excluded from the conditional leak rate database and the probability of leakage database as a result of applying these criteria are listed in Tables E-2 and E-3 of the EPRI April 22, 1994, letter. The staff has concluded that excluding the data points listed in Table E-2, with the exception of model boiler specimen 542-4 and pulled tube specimen J1-R8C74, from the 7/8-inch conditional leak rate database is appropriate. Furthermore, the staff has concluded that excluding the data points listed in Table E-3 from the 7/8-inch diameter POL database is appropriate. Pending further evaluation of the generic criteria presented in Section E.2 of the April 22, 1994 letter, the staff is continuing to assess the appropriateness of excluding data points from the conditional leak rate and POL database on a case-by-case basis.

Based on the above evaluation, the staff concludes that adequate structural and leakage integrity of the steam generator tubing can be ensured for Cycle 11 at Beaver Valley Unit 1, consistent with applicable regulatory requirements. Staff approval of this amendment request is contingent upon DLC removing at least six tube support plate intersections during this outage for destructive analysis. DLC has completed the removal of at least 6 tube support plate intersections thus satisfying the staff's requirement. In addition, the staff's approval of the proposed voltage-based repair criteria is based on DLC being able to demonstrate that the conditional probability of

burst and the primary-to-secondary leakage during a postulated MSLB will be acceptable. This report will be provided pursuant to a commitment in licensee's letter of December 13, 1994. The staff further concludes that the reduced operational leakage limits provide added assurance of steam generator tube structural and leakage integrity.

2.5 Radiological Consequences

In its evaluation of the information presented DLC including information contained in WCAP-14122, the NRC staff concluded that, based on its evaluation, the estimated primary to secondary leakage during a postulated MSLB at the end of this operating period for Beaver Valley Unit 1 is acceptable as presented by the licensee in its evaluation.

These calculations provided an estimate of the total primary to secondary leakage rate that can be expected during a postulated steam line break event, based on limiting the dose to a member of the public to a small fraction of 10 CFR Part 100 limits. The licensee concluded that a post-MSLB leakage rate of 6.5 gpm in the faulted steam generator could occur, and not exceed this small fraction of 10 CFR Part 100 limits.

Finally, DLC's evaluation considered both the preaccident iodine spike case and an event-generated iodine spike case. In this evaluation, DLC calculated site boundary thyroid doses and has concluded that the event-generated spike case is limiting, and that 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 6.5 gpm in the faulted loop. This maximum value of post-MSLB primary to secondary leakage further assumes a maximum allowable TS value of dose-equivalent I-131 of 1.0 $\mu$Ci/gm.

The NRC staff, in performing its evaluation of this event, has independently calculated the radiological consequences of a postulated steam line break, assuming a small fraction of the 10 CFR Part 100 limits at the exclusion area boundary. The licensing basis value for $\chi/Q$ of $3.1 \times 10^{-3}$ sec/m$^3$ was used in this calculation. The staff also used the dose conversion factors for iodine isotopes set forth in International Commission on Radiation Protection (ICRP) 30 as well as the breathing rates set forth in RG 1.4.

Table 1 presents the staff calculated thyroid doses for both the preaccident iodine spike case and the event-generated iodine spike case using a total projected post-MSLB primary to secondary leak rate of 6.8 gpm from all steam generators, i.e., 6.5 gpm from the faulted generator plus 0.1 gpm of normal operational leakage from each of the other generators.

**TABLE 1**

**6.8 gpm Primary to Secondary Leak Rate**

| Calculation type and acceptance criteria | Thyroid dose (rem) pre-accident spike case | Thyroid dose (rem) event-generated iodine spike |
|---|---|---|
| EAB (2 hour) | 87 | 12.3 |
| Acceptance criteria | 300 | 30 |

As can be seen from this Table, calculated thyroid doses using the licensee's limiting steam generator tube leakages are within the exposure guideline values of 10 CFR Part 100 for the preaccident iodine spike case and thus satisfy the acceptance criteria of Standard Review Plan Section 15.1.5, Appendix A, "Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR."

Similarly, calculated doses for the event-generated spike case are also shown in the above table. Calculated thyroid doses for the event-generated spike case are less than a small fraction of the exposure guideline values of 10 CFR Part 100 and satisfy the acceptance criteria.

Based on the above evaluation, the staff concludes that the radiological consequences of a MSLB outside containment for Beaver Valley Unit 1 with steam generator tube leakage up to a maximum of 6.8 gpm at the end of the operating period are acceptable. This conclusion is based on a calculated post-MSLB primary to secondary leakage not exceeding 6.8 gpm.

2.6 Summary

Based on the above evaluation, it can be concluded that adequate structural integrity of the steam generator tubing can be ensured for Cycle 11 at Beaver Valley, Unit 1 consistent with applicable regulatory requirements. In addition, the staff concludes that the methodology for determining the expected primary-to-secondary leakage during a postulated MSLB at the end of fuel Cycle 11 for Beaver Valley, Unit 1, is acceptable. The staff's approval of the proposed interim repair limit is based on the licensee being able to demonstrate that the primary-to-secondary leakage during a postulated MSLB will be acceptable. The licensee has agreed to report, prior to restart from the Refueling Outage 10, the results of the MSLB leakage analysis. The licensee has also agreed to inform the NRC prior to plant restart from the refueling outage of any unexpected inspection findings relative to the assumed characteristics of the flaws at the tube support plates. This includes any detectable circumferential indications or detectable indications outside the tube support plate thickness. The licensee's proposed changes, to revise TSs 3.4.4.5 and 4.4.4.5 and their associated Bases to allow the use of voltage-based steam generator tube plugging criteria for defects located at the tube

support plate elevations, and (2) TS 3.4.6.2 and the associated Bases of 3/4.4.6.2 to permit revised reactor coolant system operational leakage limits, are supported by the accompanying analysis and acceptable.

## 3.0 STATE CONSULTATION

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

## 4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (59 FR 42337). 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.

## 5.0 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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