

ATTACHMENT A

Beaver Valley Power Station, Unit No. 1
Proposed Technical Specification Change No. 207
MARKED-UP PAGES

The following is a list of the affected pages:

Affected Pages: 3/4 4-9
 3/4 4-10a
 3/4 4-10b
 3/4 4-10c
 3/4 4-13
 B 3/4 4-2a
 B 3/4 4-3

Attachment to "Steam Generators"

Insert "A"

- 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.

SURVEILLANCE REQUIREMENTS (Continued)

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.
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%

INSERT B

Attachment to "Steam Generators"

Insert "B"

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.

SURVEILLANCE REQUIREMENTS (Continued)

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.

INSERT C

- 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.

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.

Attachment to "Steam Generators"

Insert "C"

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.

SURVEILLANCE REQUIREMENTS (Continued)

- 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.

INSERT D

Attachment to "Steam Generators"

Insert "D"

- 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).

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

- a. No PRESSURE BOUNDARY LEAKAGE,
- b. 1 GPM UNIDENTIFIED LEAKAGE,
- c. ~~1 GPM~~ ^{450 gallons per day} total primary-to-secondary leakage through all steam generators ~~not isolated from the Reactor Coolant System and~~
¹⁵⁰ ~~500~~ gallons per day through any one steam generator, ~~not isolated from the Reactor Coolant System,~~
- d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and
- e. 28 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2230 \pm 20 psig.

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 containment atmosphere particulate and gaseous radioactivity monitor at least once per 12 hours.

BASES

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 = 500 gallons per day per steam generator). ^{Axial cracks} ~~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 500 gallons per day per steam generator can readily be detected, ¹⁵⁰ ~~by radiation monitors of steam generator~~ ~~blowdown~~. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

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.

INSERT E

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.

Attachment to "Steam Generators - Bases"

Insert "E"

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.

REACTOR COOLANT SYSTEM

BASES

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

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

3/4.4.6.2 OPERATIONAL LEAKAGE

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

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

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 28 gpm with the modulating valve in the supply line fully open at RCS pressures in excess of 2,000 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

← INSERT F

~~The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.~~

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Should PRESSURE BOUNDARY LEAKAGE occur through a component which can be isolated from the balance of the Reactor Coolant System, plant operation may continue provided the leaking component is promptly isolated from the Reactor Coolant System since isolation removes the source of potential failure.

Attachment to "Operational Leakage - Bases"

Insert "F"

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.

ATTACHMENT B

Beaver Valley Power Station, Unit No. 1 Proposed Technical Specification Change No. 207 INTERIM TUBE PLUGGING CRITERIA

A. DESCRIPTION OF AMENDMENT REQUEST

The proposed amendment would revise Specifications 3.4.5 and 3.4.6.2 including associated Bases 3/4.4.5 and 3/4.4.6.2 to allow the implementation of interim steam generator tube plugging criteria for the tube support plate elevations during Cycle 11. The allowed primary-to-secondary operational leakage from any one steam generator is being reduced from 500 gallons per day (gpd) to 150 gpd and the total allowed primary-to-secondary operational leakage through all steam generators is reduced from one gallon per minute, which is 1440 gpd, to 450 gpd.

B. BACKGROUND

Previous inservice inspections and examinations of the steam generator (SG) tubes have identified intergranular stress corrosion cracking (IGSCC) on the outer diameter of the tubes at the tube support plate (TSP) intersections. This particular form of IGSCC is known as outer diameter stress corrosion cracking (ODSCC) and is a degradation phenomenon found in a number of nuclear power plant SGs. Various tubes, including tube-to-TSP intersections, have been removed from affected SGs from numerous nuclear plants for examination and testing. Each of the pulled tubes was sectioned and metallographically examined. The examinations have revealed multiple, segmented, and axial cracks with short lengths for the deepest penetrations. The ODSCC is generally confined to within the thickness of the TSPs, consistent with the corrosion mechanism which involves the concentration of impurities, including caustics, in the tube-to-TSP crevices. There is some potential for shallow ODSCC for a short distance above or below the TSP. This has been observed in the TSP intersections of some pulled tubes from another plant.

The pulled tube specimens from Beaver Valley Unit 1 have shown only limited intergranular attack (IGA) associated with the ODSCC. However, more significant IGA has been observed to occur with ODSCC on some pulled tube specimens from other plants. These results suggest that the degradation developed as IGA plus stress corrosion cracking (SCC), particularly when maximum IGA depths greater than 25 percent are found. A large number (greater than 100) of axial cracks around the circumference are commonly found on these tubes. The maximum depth of IGA is typically one-half to one-third of the SCC depth. Patches of cellular IGA/ODSCC formed by combined axial and circumferential orientation of microcracks are frequently found in pulled tube examinations. Axial crack segments have been the dominant flaw

feature affecting the structural integrity of the pulled tube specimens as evidenced by results of burst tests of the pulled TSP intersections prior to sectioning.

C. JUSTIFICATION

Specification 4.4.5.4.a.6, Plugging or Repair Limit, requires that tubes with imperfections exceeding 40 percent of the nominal tube wall thickness be removed from service. This criterion would result in unnecessary plugging of significant numbers of SG tubes. To preclude this, interim plugging criterion (IPC) is proposed.

The IPC is based on the analysis described in WCAP-14122, "Beaver Valley Unit 1 Steam Generator Tube Plugging Criteria for Indications at Tube Support Plates," which addresses the criterion provided in draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes," to maintain SG tube serviceability. A copy of the WCAP is provided in Attachment D for NRC review and acceptance. The WCAP provides justification for a 1.0 or 2.0 volt repair limit. The methodology employed is similar to that implemented for Farley-1 and D.C. Cook-1. The proposed 1.0 volt repair limit detailed in the technical specifications reflects a conservative approach for the Beaver Valley Unit No. 1 IPC, recognizing that a 2.0 volt repair limit has been demonstrated to provide adequate margins in accordance with applicable regulatory requirements. The proposed tube plugging criteria is provided in accordance with the following:

1. A bobbin probe inspection of 100 percent of the hot leg tubes and down to the lowest tube support plate with known ODSCC at the tube support plates on the cold leg side will be performed.
2. Flaw-like signals confined within the thickness of the tube support plate with bobbin voltages less than or equal to 1.0 volt will be allowed to remain in service.
3. Flaw-like signals confined within the thickness of the tube support plate with a bobbin voltage of greater than 1.0 volt will be plugged or repaired except as noted in Item 4.
4. Flaw-like signals confined within the thickness of the tube support plate with a bobbin voltage greater than 1.0 volt but less than or equal to 3.6 volts may remain in service if a rotating pancake coil probe (RPC) inspection does not detect a flaw. Flaw indications with a bobbin voltage greater than 3.6 volts will be plugged or repaired.

5. As part of a sample inspection program to help ensure that additional degradation modes are not occurring, flaw indications with bobbin voltages greater than 1.0 volt but less than or equal to 3.6 volts will be inspected by RPC. For tubes that will be administratively plugged or repaired, no RPC inspection is required.
6. An upcoming (Cycle 11) end-of-cycle (EOC) voltage distribution will be established based upon the most recent (Cycle 10) end-of-cycle eddy current data. Based upon this distribution, postulated steam line break leakage will be estimated based on the guidance of draft NUREG-1477. Projected leakage must remain below a level which results in offsite dose estimates remaining within the licensing basis; 10 CFR 100 guidelines. For BVPS Unit 1, this level has been determined to be 6.8 gpm for all steam generators, 6.6 gpm in the limiting and assumed faulted loop. Should this estimation exceed 6.6 gpm, the highest voltage indications will be successively repaired until the leakage estimation drops below 6.6 gpm. While projected SLB leakage will be calculated as prescribed in draft NUREG-1477, Duquesne Light Company is also requesting that the current EPRI recommended leak rate calculational methodology be reviewed and approved for Cycle 11 use.
7. An overall tube burst probability during a postulated SLB event will be calculated and shown to be less than the acceptable limit defined in draft NUREG-1477 of 2.5×10^{-2} .

D. SAFETY ANALYSIS

The interim plugging criteria involves a correlation between eddy current bobbin probe signal amplitude (voltage) and indicated depth (phase angle) versus tube burst pressure and leak rate. The principal parameter is voltage amplitude which is correlated with tube burst capability and leakage potential. Indicated depth is a secondary parameter utilized as a threshold value below which added margins incorporated in the plugging criteria to minimize excessive steam line break leakage are not necessary. The plugging criteria are developed from testing of laboratory induced ODSCC specimens, extensive examination of pulled tubes from operating steam generators, and field experience from leakage due to indications at the tube support plates.

The proposed change modifies the steam generator surveillance requirements (SR) to allow implementation of the interim tube plugging criteria for Cycle 11. SR 4.4.5.2.d has been added to require 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

ODSCC indications. SR 4.4.5.4.a.6 has been modified by including an exception to the current plugging or repair limits so that the definition does not apply to the region of the tube subject to the tube support plate intersections since the interim tube plugging criteria applies to this region. SR 4.4.5.4.a.10 has been added to provide the limitations applicable for the "Tube Support Plate Interim Plugging Criteria Limit." SR 4.4.5.5.d and SR 4.4.5.5.e have been added to address additional reporting criteria for those tubes where the tube support plate IPC has been applied. Specification 3.4.6.2.c has been modified by replacing the "1 gpm" limit with a "450 gallons per day" limit, deleting "not isolated from the Reactor Coolant System" in two places, and replacing "500" with a "150" gallons per day limit through any one steam generator. The revised leakage limits are in accordance with the WCAP limitations and are consistent with the methodology addressed in draft NUREG-1477. Deleting the phrase "not isolated from the Reactor Coolant System" is consistent with the operation of the plant since SG isolation is not permitted in Modes 1, 2, 3, and 4. Bases 3/4.4.5, Steam Generators, has been modified by replacing "Cracks" with "Axial cracks," replacing "500" with "150" as the daily SG leakage limit, deleting "by radiation monitors of steam generator blowdown," and incorporating insert "E" which provides a sentence to refer to SR 4.4.5.4.a.10 for IPC criteria. Bases 3/4.4.6.2, Operational Leakage, has been modified by replacing the fourth paragraph which describes the current steam generator leakage limits with a discussion of the revised limits. The revised paragraph states that the new 150 gpd steam generator (450 gpd total) leakage limit is based on maintaining offsite doses within the CFR 100 guidelines.

In the development of the interim plugging criteria, Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" and RG 1.83 "Inservice Inspection of PWR Steam Generator Tubes" are used as the bases for determining that steam generator integrity considerations are maintained within acceptable limits. RG 1.121 describes a method, acceptable to the NRC staff, for meeting General Design Criteria 14, 15, 31, and 32. The probability and consequences of steam generator tube rupture are reduced by determining the limiting safe conditions of degradation of steam generator tubing, beyond which tubes with unacceptable cracking, as established by inservice inspection, would be removed from service by plugging. This regulatory guide uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the ASME Code. For the degradation occurring in the steam generator tube support plate elevation, tube burst criteria are inherently satisfied during normal operating conditions by the presence of the tube support plate. The presence of the tube support plate enhances the integrity of the degraded tubes in that region by precluding tube deformation beyond the diameter of the drilled hole. It is not certain whether the tube support plate would function to provide

a similar constraining effect during accident condition loadings. Therefore, no credit is taken in the development of the plugging criteria for the presence of the tube support plate during accident condition loadings. Conservatively, based on the existing data base, burst testing shows that the safety requirements for tube burst margins during both normal and accident condition loadings can be satisfied with bobbin coil signal amplitudes of about 8.82 volts or less, regardless of the depth of tube wall penetration cracking. RG 1.83 describes a method acceptable for implementing GDC 14, 15, 31, and 32 through periodic inservice inspection for detection of significant tube wall degradation.

For the interim tube plugging criteria developed for the steam generator tubes, no leakage is expected during normal operating conditions even with the presence of through wall cracks. This is the case because the stress corrosion cracking occurring in the tubes at the support plate elevations in the steam generators are short, tight, axially oriented macrocracks separated by ligaments of material. No leakage during normal operating conditions has been observed for crack indications with signal amplitudes less than 6.5 volts. Relative to the expected leakage during accident condition loadings, the limiting event with respect to primary-to-secondary leakage is a postulated steam line break event. Field and laboratory data show minimal leakage for a criterion of under 10.0 volts.

The following items support this proposed license amendment.

1. Chemistry

Duquesne Light Company has undertaken several steps to potentially help mitigate steam generator tubing corrosion. These include boric acid addition in December of 1990, removal of copper moisture separator tubes, all volatile treatment (AVT) secondary chemistry and molar ratio control according to EPRI recommendations.

2. Leak Rate Monitoring

Ability to Detect Leakage

The steam generator leak monitoring system uses N-16 detectors (RM-1MS-102A,B,C) to provide continuous indication of individual steam generator primary-to-secondary leakage. The N-16 gamma detector provides indication of steam generator leak rate in gallons per day and is used when reactor power is greater than or equal to 20 percent. Gross gamma provides a qualitative indication in counts per minute when reactor power is less than 20 percent. A recorder monitors the N-16 detectors and provides a trend recording of steam generator leak rate.

The high capacity blowdown monitor (RM-1BD-101) continuously checks the steam generator blowdown flash tank effluent. The monitor consists of a data acquisition module, sample unit, check source module, and display unit. Samples are taken from the blowdown discharged from the high capacity blowdown drain heat exchanger discharge line. The blowdown monitor provides indication and alarms locally and in the Control Room. A high-high alarm would be an indication of a primary-to-secondary leak.

The steam generator standby blowdown monitor (RM-1BD-100) continuously observes the standby steam generator blowdown tank discharge effluent. The detection system is an off-line, lead-shielded sampler. Samples are taken from blowdown discharged from the blowdown drain heat exchanger discharge line. The standby blowdown monitor produces an alarm in the Control Room only. A high-high alarm would be an indication of a primary-to-secondary leak.

The condenser air ejector vent monitor (RM-1SV-100) continuously detects gaseous effluent from the condenser air ejector vent. An alarm from this detector indicates a primary-to-secondary system leak. A high-high alarm automatically diverts the discharge of non-condensables to the containment building for subsequent manual discharge to the environment via the elevated release point.

Operator Actions for a Steam Generator Tube Leak

Abnormal radiation in a steam generator indicates primary-to-secondary leakage. This can be shown by trends or alarms on main steamline, steam jet air ejector, or SG blowdown radiation monitors, or from chemistry samples. A large leak could be indicated by feedwater flow being less than steam flow, decreasing feed flow, or decreasing feed regulating valve position in conjunction with a stable steam generator level. These symptoms, however, would more likely be noticed with a tube rupture. Procedures provide actions to mitigate a range of steam generator tube leaks up to those that can be controlled with a normal chemical and volume control system lineup.

Leak rates for large leaks (10 gpm or greater) can be determined by the operator from a charging-letdown flow balance on the RCS. Leak rates for smaller leaks use radiation levels in the secondary system and are determined by the Chemistry Department. Main steamline and steam jet air ejector radiation monitor status is trended in the main Control Room. Stable radiation monitor indications are indicative of a stable leak rate. For stable radiation monitor readings, shutdown requirements are based on absolute leak rate values. Should radiation monitor readings show a

rapid increase during a shutdown based on absolute leak rate values, the allowed time to complete the shutdown is reduced to the shortest time practical for a controlled shutdown.

Procedural Adequacy

The plant leak rate monitors and procedures provide the required indications and alarms to ensure RCS leakage is adequately detected. In addition, leakage verification is provided by our chemistry procedures which provide alternate means of calculating and confirming RCS leakage.

3. Eddy Current Test and Data Analysis Guidelines

The data analysis guidelines will be in accordance with WCAP-14122 "Appendix A" type calibration, recording, and analysis requirements.

4. Eddy Current Data Analyst Training and Qualifications

The data analyst training program will include portions on voltage measurement. A training and testing program to ensure the analysis guidelines are understood and adhered to will be given to all data analysts and monitored by an independent consultant. This same independent consultant will monitor activities throughout the job.

5. Tube Pulls

In June 1992, three tubes (2 intersections per tube) were pulled from another plant with 7/8 inch tubing in support of an IPC amendment request. Bobbin voltages of pulled tubes ranged from 1.0 to 2.02 volts, with bobbin depth predictions ranging from 19 percent to 66 percent through wall. Measured burst pressures for these intersections ranged from 9,100 to 11,200 psi.

All burst specimens had axial burst openings. The fracture faces were found to contain the deepest average crack depths. Tube R18-C21, first TSP region had the largest recorded bobbin voltage of all indications at EOC (2.02 volts), and concurrently the deepest cracking, 56 percent maximum and 42 percent average, as determined by metallography. The second TSP region of R18-C21, which had no detectable degradation (NDD) by bobbin and RPC field calls, had a maximum depth of penetration of only 38 percent, as determined by metallography. The burst pressure of this indication was 11,200 psi. The second TSP region of R18-C16 had a 1.0 volt bobbin indication, but did not exhibit a confirming RPC call. The maximum depth of penetration for the second TSP region of R18-C16 was found to 40 percent through wall, and the burst pressure was 10,200 psi.

Recently, two additional intersections from another plant with 7/8 inch tubing with relatively high voltages, 3.3 and 3.2 volts, were removed and destructively examined. The corrosion morphology of these tubes was dominated by axial ODSCC. Neither intersection contained through wall degradation. These intersections can be considered representative of the maximum EOC voltages for BVPS.

Draft NUREG-1477 states that removing tubes during each outage for examination and testing is important to enhance and validate the empirical burst and leakage correlations, to confirm that axial ODSCC continues to be the dominant degradation method at the tube support plate intersections, and to provide additional data for assessing the reliability of the inspection methods.

It is the industry position that no tube pulls are required to support the very conservative interim plugging criteria and that only tube pulls at high voltages would be necessary to support the burst and leakage correlations for full alternate repair criteria implementation. The higher voltage data would be most influential in establishing the slope and magnitude of the burst correlation at the structural limits and the probability of burst at accident condition loadings.

There is no additional need for tube pulls to support the reliability of inspection methods. The 200 intersections currently available provide an adequate database to support inspection methods and define probability of detection with adequate accuracy. Pulling tubes for refining POD would have an extremely high cost/benefit ratio as shown by the large number of NDD intersections pulled from Ringhals 3 and 4.

Based upon this data and the above discussions, no additional tube pulls are considered necessary for the next outage at BVPS.

The proposed amendment may preclude occupational radiation exposure that would otherwise be incurred by plant workers involved in tube plugging operations. It would minimize the loss of margin in the reactor coolant flow through the steam generator in LOCA analyses. The proposed amendment would avoid loss of margin in reactor coolant system flow and, therefore, assist in demonstrating that minimum flow rates are maintained in excess of that required for operation at full power. Reduction in the amount of tube plugging required can reduce the length of plant outages and reduce the time that the steam generator is open to the containment environment during an outage. Based on the methodology described in the WCAP, DLC has determined that this methodology is applicable to our steam generators and provides a safe and effective alternative to plugging.

E. NO SIGNIFICANT HAZARDS EVALUATION

The no significant hazard considerations involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
- (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
- (3) Involve a significant reduction in a margin of safety.

The following evaluation is provided for the no significant hazards consideration standards.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Testing of model boiler specimens for free span tubing (no tube support plate [TSP] restraint) at room temperature conditions show burst pressures in excess of 5000 psi for indications of outer diameter stress corrosion cracking with voltage measurement as high as 19 volts. Burst testing performed on intersections pulled from BVPS with up to a 2.7 volt indication shows measured burst pressure in excess of 6600 psi at room temperature. Burst testing performed on pulled tubes from other plants with up to 7.5 volt indications show burst pressures in excess of 6300 psi at room temperatures. Correcting for the effects of temperature on material properties and minimum strength levels (as the burst testing was done at room temperature), tube burst capability significantly exceeds the safety factor requirements of RG 1.121. As stated earlier, tube burst criteria are inherently satisfied during normal operating conditions due to the proximity of the TSP. Test data indicates that tube burst cannot occur within the TSP, even for tubes which have 100 percent through wall electric discharge machining (EDM) notches, 0.75 inch long, provided that the TSP is adjacent to the notched area. Since tube to TSP proximity precludes tube burst during normal operating conditions, use of the criteria must retain tube integrity

characteristics which maintain a margin of safety of 1.43 times the bounding faulted condition steam line break (SLB) pressure differential. As previously stated, the RG 1.121 criterion requiring maintenance of a safety factor of 1.43 times the SLB pressure differential on tube burst is satisfied by 7/8 inch diameter tubing with bobbin coil indications with signal amplitudes less than 8.82 volts regardless of the indicated depth measurement. The plugging criteria (resulting in a projected end-of-cycle [EOC] voltage) compares favorably with the 8.82 volt structural limit considering the extremely slow apparent voltage growth rate of indications at BVPS. Using the established methodology of RG 1.121, the structural limit is reduced by allowances for uncertainty and growth to develop a beginning-of-cycle (BOC) repair limit which should preclude indications at EOC conditions which exceed the structural limit. The non-destructive examination (NDE) uncertainty component is 20.5 percent and is based on the EPRI Alternate Repair Criteria (ARC). A bounding growth allowance of 40 percent will be applied. This value is conservative for BVPS Unit 1. The BOC maximum allowable repair limit should not permit the existence of EOC indications (when the 40 percent growth and 20.5 percent uncertainty allowances are applied) which exceed the 8.82 volt structural limit. By adding NDE uncertainty allowances and an allowance for crack growth to the repair limit, the structural limit can be validated. Therefore, the maximum allowable BOC repair limit (RL) based on the structural limit of 8.82 volts can be represented by the expression:

$$RL + (0.205 \times RL) + (0.40 \times RL) = 8.82 \text{ volts, or}$$

the maximum allowable BOC repair limit can be expressed as:

$$RL = 8.82 \text{ volt structural limit} / 1.605 = 5.5 \text{ volts.}$$

It is reasonable that this repair limit (5.5 volts) could be applied for IPC implementation to repair bobbin indications greater than 1.0 or 2.0 volts independent of RPC confirmation of the indication. The analyses were performed based on a 1.0 or 2.0 volt repair limit. Duquesne Light Company has chosen to use a steam generator tube repair limit of 1.0 volt. Conservatively, an upper limit of 3.6 volts will be used to assess tube integrity for those bobbin indications which are above 1.0 volt but do not have confirming RPC calls. This 3.6 volt upper limit for non-confirmed RPC calls is consistent with other recently approved IPC programs for the two other plants with 7/8 inch tubing that currently implement IPCs. Since the upper bound for repair of non-confirmed RPC is limited to a value far less than the limit associated with a full alternate criteria, the establishment of the repair limits are judged to be independent of the pulled tube data base used.

The conservatism of the growth allowance used to develop the repair limit is shown by the most recent BVPS eddy current data. The average voltage growth for all indications was 16 percent while the average voltage growth for indications greater than 0.75 volts at BOC was 6 percent. The largest overall voltage growth in a particular steam generator was found in the "A" steam generator, which had an overall average growth of 25 percent. Only two tubes had an absolute voltage growth which exceeded 1.0 volt for Cycle 9. The maximum absolute voltage growth in the 1993 inspection was recorded to be 1.18 volts. Each of the last three inspections, which included 100 percent of all hot leg tubes, showed decreasing voltage growth trends in each successive inspection for all categories; overall voltage growth, growth of BOC indications less than 0.75 volts, and growth of indications greater than 0.75 volts. The decreasing voltage growth rate trend data indicates that DLC has good control of the ODSCC occurring in the BVPS Unit 1 steam generators and also implies that atypical voltage growth of a few indications is unlikely.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated main SLB outside of containment but upstream of the main steam isolation valve (MSIV) represents the most limiting radiological condition relative to the IPC. In support of implementation of the interim plugging criteria, it will be determined whether the distribution of cracking indications at the TSP intersections at the end of Cycle 11 are projected to be such that primary-to-secondary leakage would result in site boundary doses within a small fraction of the 10 CFR 100 guidelines. A separate calculation has determined this allowable SLB leakage limit to be 6.6 gpm in the faulted loop. This limit was calculated using the Technical Specification RCS Iodine-131 activity level of 1.0 micro Curies per gram dose equivalent Iodine-131 and the recommended Iodine-131 transient spiking values consistent with NUREG-0800. The projected SLB leakage rate calculation methodology prescribed in Section 3.3 of draft NUREG-1477 will be used to calculate EOC leakage. The log-logistic probability of leakage correlation will be used to establish the SLB leak rate used for comparison with the 6.6 gpm faulted loop allowable limit. Due to the relatively low voltage levels of indications at BVPS and low voltage growth rates, it is expected that the actual calculated leakage values will be far less than this limit. Additionally, the current Iodine-131 levels as of May 1994 at BVPS are about 1000 times less than the Technical Specification limit of 1.0.

Application of the criteria requires the projection of postulated SLB leakage, based on the projected EOC voltage distribution for the upcoming cycle. Projected EOC voltage

distribution is developed using the most recent EOC eddy current results and a voltage measurement uncertainty. Data indicate that a threshold voltage of 2.8 volts would result in through wall cracks long enough to leak at steam line break conditions. Draft NUREG-1477 requires that all indications to which the IPC are applied must be included in the leakage projection. Tube pull results from another plant with 7/8 inch tubing with a substantial voltage growth data base have shown that tube wall degradation of greater than 40 percent through wall was readily detectable either by the bobbin or RPC probe. The tube with maximum through wall penetration of 56 percent (42 percent average) had a voltage of 2.02 volts. This indication also was the largest recorded bobbin voltage from the EOC eddy current data. Based on the BVPS pulled tube and industry pulled tube data supporting a lower threshold for SLB leakage of 2.8 volts, inclusion of all IPC intersections in the leakage model is quite conservative. The ODSCC occurring at BVPS has historically resulted in relatively low voltage levels and has exhibited decreasing voltage growth trends over the last three inspections. BVPS has not identified ODSCC as a contributor to operational leakage. The current leakage levels at BVPS are negligible (less than 1 gpd). In order to satisfy the requirements of draft NUREG-1477, EOC 10 eddy current data will be used to calculate the projected SLB leakage according to draft NUREG-1477 methodology. Leakage calculated using the recommended EPRI leakage correlation will also be provided. Duquesne Light Company is requesting that the NRC review and approve the EPRI SLB leakage calculation methodology. Sufficient justification is included to establish acceptability of the EPRI leakage correlation based on criteria provided by the NRC in the February 8, 1994, Industry/NRC working meeting on the voltage based criteria.

In order to assess the sensitivity of application of the voltage based criteria upon SLB leakage, the EOC 9 eddy current results were used to calculate postulated EOC 10 leakage using both the NUREG-1477 methodology and EPRI correlation assuming that a 1.0 or 2.0 volt plugging limit were implemented at the BOC 10. Results indicate SLB leakage of 0.46 gpm and 0.044 gpm using the NUREG and EPRI methodologies with an assumed probability of detection (POD) of 0.6 for a 2.0 volt repair limit. Since Duquesne Light Company has chosen to limit the voltage based plugging limit at 1.0 volt, EOC 11 SLB leakage is analyzed to be approximately 5 percent lower than the calculated SLB leakage with a 2.0 volt repair limit.

Therefore, implementation of the interim plugging criteria does not adversely affect steam generator tube integrity and implementation will be shown to result in acceptable dose consequences, therefore, the proposed amendment does not

result in any increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

Implementation of the proposed steam generator tube interim TSP plugging criteria does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in an accident outside of the region of the TSP elevations; no ODSCC that has been identified at the TSP has been detected outside the thickness of the TSPs. Neither a single or multiple tube rupture event would be expected in a steam generator in which the plugging criteria has been applied (during all plant conditions).

Specifically, Duquesne Light Company will implement a maximum leakage rate limit of 150 gpd per steam generator to help preclude the potential for excessive leakage during all plant conditions. The technical specification limits on primary-to-secondary leakage at operating conditions are to be a maximum of 450 gpd for all steam generators, or, a maximum of 150 gpd for any one steam generator. The RG 1.121 criterion for establishing operational leakage rate limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture during faulted plant conditions. The 150 gpd limit should provide for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. RG 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before-break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded.

The single through wall crack lengths that result in tube burst at 1.43 times the steam line break pressure differential and SLB pressure differential alone are approximately 0.57 inch and 0.84 inch, respectively. A leak rate of 150 gpd will provide for detection of 0.41 inch long cracks at nominal leak rates and 0.62 inch long cracks at the lower 95 percent confidence level leak rates. Since tube burst is precluded during normal operation due to the proximity of the TSP to the tube and the potential exists for the crevice to become uncovered during SLB conditions, the leakage from the maximum permissible crack must preclude tube burst at SLB conditions. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for SLB conditions using the lower 95 percent leakage data. Additionally, this leak-before-break evaluation assumes that

the entire crevice area is uncovered during blowdown. Partial uncovering will provide benefit to the burst capacity of the intersection. Analyses have shown that only a small percentage of the TSPs are deflected greater than the TSP thickness during a postulated SLB.

Steam generator tube integrity continues to be maintained through inservice inspection and primary-to-secondary leakage monitoring, therefore, the possibility of a new or different kind of accident from any accident previously developed is not created.

3. Does the change involve a significant reduction in a margin of safety?

The use of the voltage based bobbin probe interim TSP elevation plugging criteria is demonstrated to maintain steam generator tube integrity commensurate with the requirements of RG 1.121. RG 1.121 describes a method acceptable to the NRC staff for meeting GDCs 14, 15, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criteria, even under the worst case conditions, the occurrence of ODSCC at the TSP elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The EOC distribution of crack indications at the TSP elevations will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions and that radiological consequences are not adversely impacted.

In addressing the combined effects of loss of coolant accident (LOCA) and safe shutdown earthquake (SSE) on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case as the TSP may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the RC² flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase peak clad temperature (PCT). Second, there is a potential that partial through wall cracks in tubes could progress to through wall cracks during tube deformation or collapse.

Consequently, since the leak-before-break methodology is applicable to the BVPS reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. The limiting LOCA event becomes either the accumulator line break or the pressurizer surge line break. LOCA loads for the primary pipe breaks were used to bound the conditions at BVPS for smaller breaks. The results of the analysis using the larger break inputs show that the LOCA loads were found to be of insufficient magnitude to result in steam generator tube collapse or significant deformation. The LOCA and SSE tube collapse evaluation performed for another plant with Series 51 steam generators using bounding input conditions (large break loadings) is considered applicable to BVPS.

Addressing RG 1.83 considerations, implementation of the bobbin probe voltage based interim tube plugging criteria is supplemented by: enhanced eddy current inspection guidelines to provide consistency in voltage normalization, a 100 percent eddy current inspection sample size at the TSP elevations, and RPC inspection requirements for the larger indications left inservice to characterize the principal degradation as ODSCC.

As noted previously, implementation of the TSP elevation plugging criteria will decrease the number of tubes which must be repaired. The installation of steam generator tube plugs reduces the RCS flow margin. Thus, implementation of the alternate plugging criteria will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to plant safety as defined in the Final Safety Analysis Report or any Bases of the Technical Specifications.

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the considerations expressed above, it is concluded that the activities associated with this license amendment request satisfies the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.