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U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Beaver Valley Power Station, Unit No. 1
BV-1 Docket No. 50-334, License No. DPR-66
Proposed Operating License Change Request No. 207
Interim Steam Generator Tube Plugging Criteria

Pursuant to 10 CFR 50.90, Duquesne Light Company requests an amendment to the above license in the form of changes to the technical specifications. The proposed changes will incorporate interim steam generator tube plugging criteria.

The proposed technical specification changes are presented in Attachment A. The safety analysis (including the no significant hazards evaluation) is presented in Attachment B. The typed replacement technical specification pages are presented in Attachment C. A Westinghouse application for withholding proprietary information along with proprietary WCAP-14122 and non-proprietary WCAP-14123, "Beaver Valley Unit 1 Steam Generator Tube Plugging Criteria for Indications at Tube Support Plates," are presented in Attachment D.

As WCAP-14122 contains information proprietary to Westinghouse Electric Corporation, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations. Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-94-661 and should be addressed to Mr. Nicholas J. Liparulo, Manager, Nuclear Safety Regulatory and Licensing Activities, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

This change has been reviewed by the Onsite Safety Committee and Offsite Review Committee. The change was determined to be safe and does not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the attached safety analysis.

Change: NRC PDR 1 Ltr Encl. A001 1 INP

ATTACHMENT C

Beaver Valley Power Station, Unit No. 1
Proposed Technical Specification Change No. 207
TYPED PAGES

Applicable Typed Pages

ATTACHMENT TO LICENSE AMENDMENT NO. ______

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 4-13	3/4 4-13
B 3/4 4-2a	B 3/4 4-2a
B 3/4 4-3	B 3/4 4-3
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5
B 3/4 4-6	B 3/4 4-6

(Proposed Wording)

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 inservice 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).

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. 450 gallons per day total primary-to-secondary leakage through all steam generators and 150 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 28 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2230 ±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.

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.

3/4.4.6.1 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.

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.

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

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

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.

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.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chmistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity $>$ 1.0 μCi/gram DOSE, EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limits shown on Figure 3.4-1 must be restricted since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing Tavg to $<$ 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.4 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal-induced

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves, Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the time in life indicated on the respective curves.

The reactor vessel materials have been tested to determine their initial RT_{NTD}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NTD}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using Figures B 3/4.4-1 and B 3/4.4-2. The heatup and cooldown limit curves, Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NTD}.

The heatup and cooldown curves have been developed in accordance with the methodology provided in Regulatory Guide 1.99, Revision 2 and no longer contain the additional margin of 10°F and 60 psig for instrument error previously incorporated in these curves.

ATTACHMENT D

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

The following are enclosed:

Westinghouse Application for Withholding Proprietary Information

Proprietary WCAP-14122
and
Non-Proprietary WCAP-14123

"Beaver Valley Unit 1 Steam Generator
Tube Plugging Criteria for Indications at Tube Support Plates"