50-412

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

August 18, 1999

Mr. J. E. Cross
President-Generation Group
Duquesne Light Company
Post Office Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: ALTERNATE PLUGGING CRITERIA FOR STEAM GENERATOR TUBING AND REDUCTION OF REACTOR COOLANT SYSTEM SPECIFIC ACTIVITY LIMITS (TAC NOS. M95793 AND M99828)

Dear Mr. Cross:

The Commission has issued the enclosed Amendment No. 101 to Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit 2 (BVPS-2). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated June 18, 1996, as supplemented December 12, 1997, February 23, June 15, and July 15, 1999, which submitted Proposed Operating License Change Request (LCR) Nos. 235 and 109. Proposed Operating Change Request No. 235, applicable to Beaver Valley Power Station, Unit No. 1 (BVPS-1), was previously issued as Amendment No. 219 to Facility Operating License No. DPR-66. Additionally, this amendment consists of changes to the TSs in response to your application dated October 22, 1997, as supplemented February 23, June 28, and July 15, 1999, which submitted Proposed Operating LCR No. 115.

This amendment implements:

- a voltage-based repair criteria for BVPS-2 steam generator tubes similar to the changes approved for BVPS-1 in License Amendment No. 198. The changes revise BVPS-2 TSs 4.4.5 and 3.4.6.2 and associated Bases to reflect the guidance provided in the Nuclear Regulatory Commission's (NRC) Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," (GL 95-05). Additionally, BVPS-2 TS Table 4.4-2 is revised to reference TS 6.6 for reporting requirements.

- reduced reactor coolant system specific activity limits in accordance with the NRC's guidance provided in GL 95-05. The definition of DOSE EQUIVALENT I-131 is replaced with the Improved Standard TS definition in the first sentence, and an equation is added based on dose conversion derived from the International Commission on Radiation Protection (ICRP) ICRP-30. TS 3.4.8, Specific Activity, is revised by reducing the DOSE EQUIVALENT I-131 limit from 1.0 μCi/gram to 0.35 μCi/gram for the 48-hour limit and from 60 μCi/gram to 21 μCi/gram for the maximum instantaneous limit. Item 4.a in TS Table 4.4-12, Primary Coolant Specific Activity Sample and Analysis Program, TS Figure 3.4-1, and the Bases for TS 3/4.4.8 are also modified to reflect the reduced DOSE EQUIVALENT I-131 limit.

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Your letter dated February 23, 1999, provided a revised control room dose calculation in support of both LCR Nos. 109 and 115 for BVPS-2. Importantly, this calculation assumed the lower allowable primary-to-secondary leak rate limits which would be implemented by LCR 109, and the reduced RCS specific activity limits which would be implemented by LCR 115. Because of this interdependence, the changes of LCR 109 must be implemented concurrently with those of LCR 115 in order for the supporting analysis to remain valid. Hence, LCRs 109 and 115 have been combined into this single amendment.

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

Sincerely,

Victor Nerses for

Daniel S. Collins, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-412

Enclosures: 1. Amendment No. 101 to NPF-73
2. Safety Evaluation

cc w/encls: See next page

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Sincerely,

Original signed by: V. Nerses for

Daniel S. Collins, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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(originally gave NLO w/comments on 8/11/99.)

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

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 101
License No. NPF-73

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

   A. The application for amendment by Duquesne Light Company, et al. (DLCo; the licensee) dated June 18, 1996, as supplemented December 12, 1997, February 23, June 15, and July 15, 1999; and the application for amendment dated October 22, 1997, as supplemented February 23, June 28, and July 15, 1999, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, 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. NPF-73 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 101 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. DLCO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

S. Singh Bajwa, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 18, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 101

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

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

Remove	Insert
1-4	1-4
1-5	1-5
1-6	1-6
1-7	1-7
3/4 4-12	3/4 4-12
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
3/4 4-14a	3/4 4-14a
3/4 4-14b	3/4 4-14b
- - - -	3/4 4-14c
- - - -	3/4 4-14d
- - - -	3/4 4-14e
- - - -	3/4 4-14f
3/4 4-16	3/4 4-16
3/4 4-19	3/4 4-19
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
3/4 4-29	3/4 4-29
B 3/4 4-3	B 3/4 4-3
B 3/4 4-3a	B 3/4 4-3a
- - - -	B 3/4 4-3b
B 3/4 4-4e	B 3/4 4-4e
B 3/4 4-4f	B 3/4 4-4f
B 3/4 4-4g	B 3/4 4-4g
B 3/4 4-4h	B 3/4 4-4h
B 3/4 4-4i	B 3/4 4-4i
B 3/4 4-4j	B 3/4 4-4j
- - - -	B 3/4 4-4k
B 3/4 4-5	B 3/4 4-5
B 3/4 4-6	B 3/4 4-6
B 3/4 7-2j	B 3/4 7-2j

1.15 THROUGH 1.17 (DELETED)

QUADRANT POWER TILT RATIO (QPTR)

1.18 QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The DOSE EQUIVALENT I-131 is calculated with the following equation:

$$C_{I-131_{D.E.}} = C_{I-131} + \frac{C_{I-132}}{170} + \frac{C_{I-133}}{6} + \frac{C_{I-134}}{1000} + \frac{C_{I-135}}{34}$$

Where "C" is the concentration, in microcuries/gram of the iodine isotopes. This equation is based on dose conversion factors derived from ICRP-30.

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals;

b. The testing of one (1) system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

REACTOR TRIP SYSTEM RESPONSE TIME

1.22 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.23 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

AXIAL FLUX DIFFERENCE

1.24 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two-section excore neutron detector.

PHYSICS TESTS

1.25 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.26 $\bar{E}$ shall be the average sum (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

SOURCE CHECK

1.27 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

PROCESS CONTROL PROGRAM

1.28 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

1.29 DELETED

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.30 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.6 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications in the Administrative Control section.

GASEOUS RADWASTE TREATMENT SYSTEM

1.31 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting Primary Coolant System offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

VENTILATION EXHAUST TREATMENT SYSTEM

1.32 VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

PURGE-PURGING

1.33 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions, in such a manner that replacement air or gas is required to purify the confinement.

VENTING

1.34 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

MAJOR CHANGES

1.35 MAJOR CHANGES to radioactive waste systems (liquid, gaseous and solid), as addressed in the PROCESS CONTROL PROGRAM, shall include the following:

1) MAJOR CHANGES in process equipment, components, structures, and effluent monitoring instrumentation from those described in the Final Safety Analysis Report (FSAR) or the Hazards Summary Report and evaluated in the staff's Safety Evaluation Report (SER) (e.g., deletion of evaporators and installation of demineralizers; use of fluidized bed calciner/incineration in place of cement solidification systems);

2) MAJOR CHANGES in the design of radwaste treatment systems (liquid, gaseous, and solid) that could significantly increase the quantities or activity of effluents released or volumes of solid waste stored or shipped offsite from those previously considered in the FSAR and SER (e.g., use of asphalt system in place of cement);

3) Changes in system design which may invalidate the accident analysis as described in the SER (e.g., changes in tank capacity that would alter the curies released); and

4) Changes in system design that could potentially result in a significant increase in occupational exposure of operating personnel (e.g., use of temporary equipment without adequate shielding provisions).

MEMBER(S) OF THE PUBLIC

1.36 MEMBER(S) OF THE PUBLIC means any individual except when that individual is receiving an occupational dose.

CORE OPERATING LIMITS REPORT

1.37 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.5. Plant operation within these operating limits is addressed in individual specifications.

SURVEILLANCE REQUIREMENTS (Continued)

1. All nonplugged tubes that previously had detectable wall penetrations greater than 20 percent, and

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.

5. Indications left in service as a result of application of the tube support plate voltage-based repair criteria (4.4.5.4.a.10) shall be inspected by bobbin coil probe during all future refueling outages.

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. Implementation of the steam generator tube-to-tube support plate repair criteria requires a 100-percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.

SURVEILLANCE REQUIREMENTS (Continued)

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

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 All Volatile Treatment (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.

SURVEILLANCE REQUIREMENTS (Continued)

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 increased to at least once per 20 months. The increase 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 steamline 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.

SURVEILLANCE REQUIREMENTS (Continued)

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%

      This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to 4.4.5.4.a.10 for the repair limit applicable to these intersections.

   b) ABB Combustion Engineering TIG welded sleeve wall 32%

   c) Westinghouse laser welded sleeve wall 25%

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 steamline 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.

SURVEILLANCE REQUIREMENTS (Continued)

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) ABB Combustion Engineering TIG welded sleeves, CEN-629-P, Revision 02 and CEN-629-P Addendum 1.

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

10. Tube Support Plate Plugging Limit is used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging (repair) limit is based on maintaining steam generator tube serviceability as described below:

   a) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to 2.0 volts will be allowed to remain in service.

   b) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts will be repaired or plugged, except as noted in 4.4.5.4.a.10.c below.

SURVEILLANCE REQUIREMENTS (Continued)

c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to the upper voltage repair limit[1] may remain in service if a rotating pancake coil or acceptable alternative inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit[1] will be plugged or repaired.

d) If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.a.10.a, 4.4.5.4.a.10.b, and 4.4.5.4.a.10.c.

The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL})\left(\frac{CL - \Delta t}{CL}\right)$$

(1) The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

SURVEILLANCE REQUIREMENTS (Continued)

where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage repair limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle
Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth rate per cycle length
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20-percent has been approved by NRC)[2]

Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.4.5.4.a.10.a, 4.4.5.4.a.10.b, and 4.4.5.4.a.10.c.

(2) The NDE is the value provided by the NRC in GL 95-05 as supplemented.

SURVEILLANCE REQUIREMENTS (Continued)

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 submitted in a Special Report in accordance with 10 CFR 50.4.

b. The complete results of the steam generator tube and sleeve inservice inspection shall be submitted in a Special Report in accordance with 10 CFR 50.4 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 implementation of the voltage-based repair criteria to tube support plate intersections, notify the Commission prior to returning the steam generators to service (MODE 4) should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.

SURVEILLANCE REQUIREMENTS (Continued)

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines of the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the Commission and provide an assessment of the safety significance of the occurrence.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug or repair defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to 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

$S = \frac{9}{n}\%$ Where 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. 150 gallons per day primary-to-secondary LEAKAGE through any one steam generator, and

d. 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.

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. ≤ 0.35 μCi/gram DOSE EQUIVALENT I-131, and

b. ≤ 100/$\bar{E}$ μCi/gram

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

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the primary coolant > 0.35 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in HOT STANDBY with T_{avg} < 500°F within 6 hours.

b. With the specific activity of the primary coolant > 100/$\bar{E}$ μCi/gram, be in HOT STANDBY with T_{avg} < 500°F within 6 hours.

MODES 1, 2, 3, 4, and 5

a. With the specific activity of the primary coolant > 0.35 μCi/gram DOSE EQUIVALENT I-131 or > 100/$\bar{E}$ μCi/gram, perform the sampling analysis requirement of item 4a of Table 4.4-12 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the primary coolant shall be determined to be within the performance limits of the sampling and analysis program of Table 4.4-12.

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* With T_{avg} ≥ 500°F.

TABLE 4.4-12

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY	MODES IN WHICH SURVEILLANCE REQUIRED
1. Gross Activity Determination	3 times per 7 days with a maximum time of 72 hours between samples.	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1,
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months	1,
4. Isotopic Analysis for Iodine including I-131 I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 0.35 µCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ µCi/gram, and	1#,2#,3#,4#,5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

#Until the specific activity of the primary coolant system is restored to within its limits.

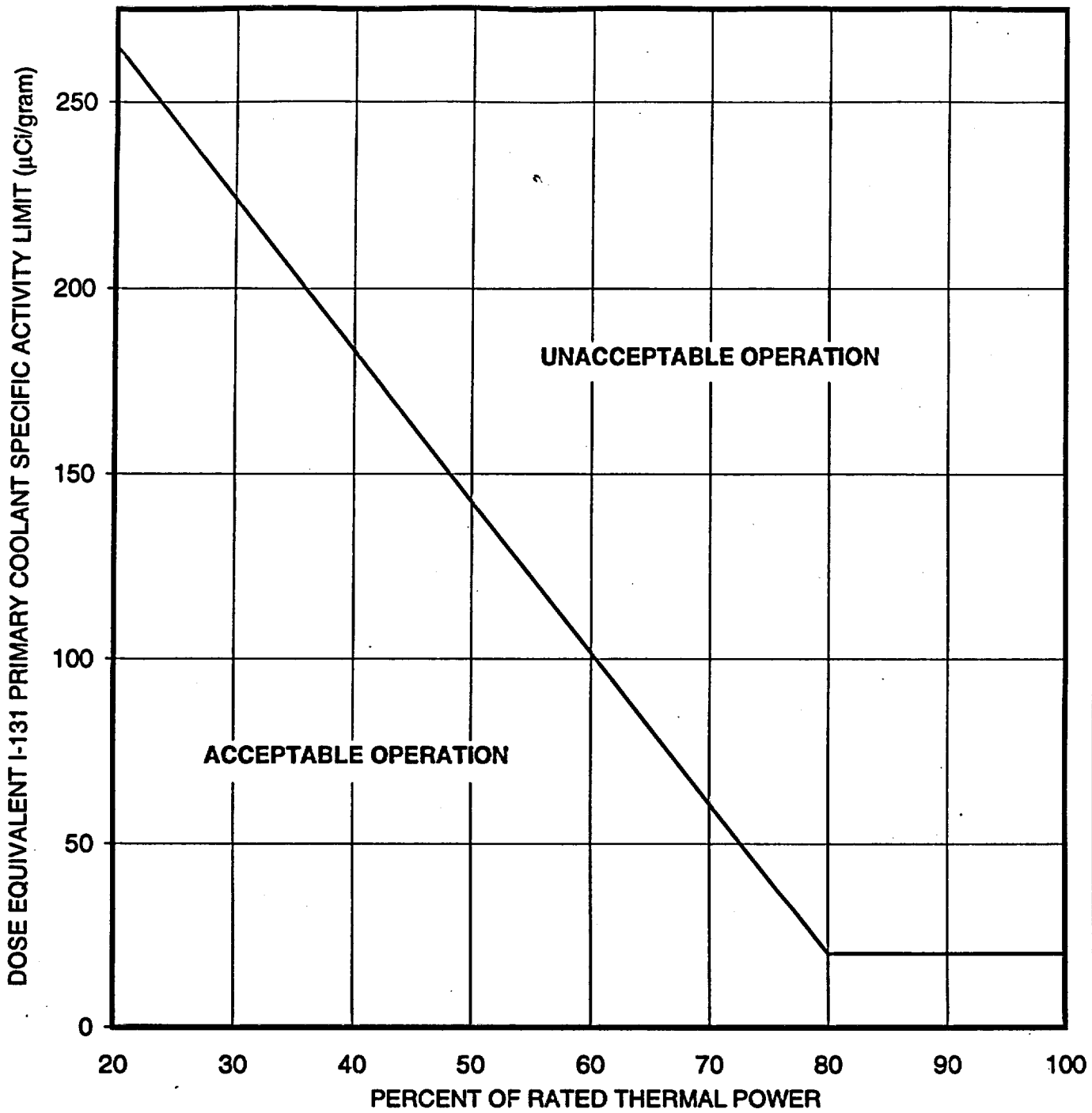

FIGURE 3.4-1

DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 0.35 µCi/gram Dose Equivalent I-131

3/4.4.5 STEAM GENERATORS (Continued)

decay heat removal capabilities for RCS temperatures greater than 350°F if one steam generator becomes inoperable due to single failure considerations. Below 350°F, decay heat is removed by the RHR system.

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the 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 or repaired by sleeving. The technical bases for sleeving are described in the approved vendor reports listed in Surveillance Requirement 4.4.5.4.a.9.

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

3/4.4.5 STEAM GENERATORS (Continued)

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.

The voltage-based repair limits of these surveillance requirements (SR) implement the guidance in GL 95-05 and are applicable only to Westinghouse-designed steam generators (SGs) with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The guidance in GL 95-05 will not be applied to the tube-to-flow distribution baffle plate intersections. The voltage-based repair limits are not applicable to other forms of SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no NDE detectable cracks extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of these SRs requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation, at the 95-percent prediction interval curve reduced to account for the lower 95/95-percent tolerance bound for tubing material properties at 650°F (i.e., the 95-percent LTL curve). The voltage structural limit must be adjusted downward to account for potential degradation growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit; V_{URL}, is determined from the structural voltage limit by applying the following equation:

$$V_{URL} = V_{SL} - V_{Gr} - V_{NDE}$$

3/4.4.5 STEAM GENERATORS (Continued)

where V_{Gr} represents the allowance for degradation growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in GL 95-05.

Safety analyses were performed pursuant to Generic Letter 95-05 to determine the maximum MSLB-induced primary-to-secondary leak rate that could occur without offsite doses exceeding a small fraction of 10 CFR 100 (concurrent iodine spike), 10 CFR 100 (pre-accident iodine spike), and without control room doses exceeding GDC-19. The current value of the maximum MSLB-induced leak rate and a summary of the analyses are provided in Section 15.1.5 of the UFSAR.

The mid-cycle equation in SR 4.4.5.4.a.10.d should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates.

SR 4.4.5.5 implements several reporting requirements recommended by GL 95-05 for situations which the NRC wants to be notified prior to returning the SGs to service. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle (EOC) voltage distribution (refer to GL 95-05 for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the SGs to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing the GL section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per the GL section 6.b (c) criteria.

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)

BACKGROUND (Continued)

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30, requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100 percent leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

Except for primary-to-secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary-to-secondary LEAKAGE as the initial condition. An exception to the primary-to-secondary LEAKAGE is described below for the main steamline break (MSLB) analyzed in support of voltage-based steam generator tube repair criteria.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

APPLICABLE SAFETY ANALYSES (Continued)

Primary-to-secondary LEAKAGE is a factor in the dose releases outside containment resulting from a MSLB accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The MSLB is more limiting for site radiation releases. The primary-to-secondary LEAKAGE assumed in the safety analysis for the MSLB accident is described in UFSAR Section 15.1.5. The radiological consequences of a MSLB outside of containment was reanalyzed in support of the tube support plate voltage-based repair criteria stated in SR 4.4.5.4.a.10. For this analysis, the thyroid dose was maximized at 10% of the 10 CFR Part 100 guideline of 300 rem for the co-incident iodine spike case. RCS leakage was based on projection rather than on technical specification leakage limits. The analysis indicated that offsite doses would remain within regulatory criteria with the assumed primary-to-secondary leakage (described in UFSAR Section 15.1.5) should steam generator tubes fail due to the depressurization associated with a MSLB.

A similar analysis was performed using a control room thyroid dose of 30 rem as the criterion. The control room was assumed to be manually isolated and pressurized at T=30 minutes for a period of one hour, at which time filtered emergency intake would be automatically started. The control room would be purged with fresh air at T=8 hours following release cessation. The analysis indicated that control room doses would remain within regulatory criteria with the assumed primary-to-secondary leakage (described in UFSAR Section 15.1.5) should steam generator tubes fail due to the depressurization associated with a MSLB.

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Should pressure boundary LEAKAGE occur through a

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

LCO (Continued)

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.

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.

c. Primary-to-Secondary LEAKAGE through Any One SG

Operating experience at PWR plants has shown that sudden increases in leak rate are often precursors to larger tube failures. Maintaining an operating LEAKAGE limit of 150 gpd per steam generator will minimize the potential for a large LEAKAGE event at power. This operating LEAKAGE limit is more restrictive than the operating LEAKAGE limit in standardized technical specifications. This provides additional margin to accommodate a tube flaw which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. This reduced LEAKAGE limit, in conjunction with a leak rate monitoring program, provides additional assurance that this precursor LEAKAGE will be detected and the plant shut down in a timely manner.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

LCO (Continued)

d. 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 3.4.6.2, "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.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

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.

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.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

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.

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.

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 chemistry 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 primary coolant specific activity is limited in order to maintain offsite and control room operator doses associated with postulated accidents within applicable requirements. Specifically, the 0.35 µCi/gm DOSE EQUIVALENT I-131 limit ensures that the offsite dose does not exceed a small fraction of 10 CFR Part 100 guidelines and that control room operator thyroid dose does not exceed GDC-19 in the event of primary-to-secondary leakage induced by a main steam line break.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 0.35 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 0.35 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 to ensure that assumptions made in the UFSAR accident analyses are not exceeded.

Reducing T_{avg} to < 500°F minimizes 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

3/4.4.8 SPECIFIC ACTIVITY (Continued)

relief valves. This action also reduces the pressure differential across the steam generator tubes reducing the probability and magnitude of main steam line break accident induced primary-to-secondary leakage. 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 3.9 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 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.

BASES

3/4.7.1.3 PRIMARY PLANT DEMINERALIZED WATER (PPDW)

The OPERABILITY of the PPDW storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to atmosphere.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that steam releases to the environment will not be significant contributors to radioactivity releases resulting from analyzed accidents. Many of the analyzed accidents assume that a loss of auxiliary AC power occurs, making the main condenser unavailable for plant cooldown, and making it necessary to dump steam to the environment via SG atmospheric dump valves. Maintaining secondary system specific activity within the limits ensures that these releases, in conjunction with other releases associated with the accident, will be within applicable dose criteria.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. NPF-73

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNIT 2

DOCKET NO. 50-412

1.0 INTRODUCTION

By letter dated June 18, 1996, as supplemented December 12, 1997, February 23, June 15, and July 15, 1999, Duquesne Light Company (DLC; the licensee), submitted a request for changes to the technical specifications (TSs) for Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and BVPS-2). For BVPS-1, only editorial changes were proposed. Those editorial changes were previously approved by Amendment No. 219 to Facility Operating License No. DPR-66 for BVPS-1. For BVPS-2, the licensee proposed to implement voltage-based alternate repair criteria for steam generator tubes. The proposed alternate repair criteria would allow steam generator tubes having outside diameter stress corrosion cracking (ODSCC) that is predominately axially oriented and confined within the tube support plates (TSPs) to remain in service on the basis of bobbin coil voltage response. The Nuclear Regulatory Commission (NRC) guidance on the alternate repair criteria is specified in Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," (GL 95-05). The licensee also requested that BVPS-2 be approved to operate based upon a 3.0 gallons-per-minute (gpm) primary-to-secondary leak initiated by an accident in the faulted steam generator in addition to the TS allowable value for a primary-to-secondary leak from each of the two intact steam generators of 150 gallons-per-day (gpd) per steam generator. BVPS-1 has already implemented voltage-based alternate repair criteria for steam generator tubes by Amendment No. 198 to Facility Operating License No. DPR-66. The December 12, 1997, letter providing clarifying information; the February 23, 1999, letter providing a revised control room dose calculation; and the June 15, and July 15, 1999, letters providing final copies of the revised TS pages, did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the November 18, 1998, Federal Register notice. In addition, by a letter dated February 13, 1997, which was not associated with these submittals, the licensee requested approval to use a voltage-dependent probability of detection for application in steam generator voltage-based alternate repair criteria for both units.

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By letter dated October 22, 1997, as supplemented February 23, June 28, and July 15, 1999, DLC submitted a separate request for changes to the TSs for BVPS-2. The requested changes would modify the TSs by reducing the reactor coolant system (RCS) specific activity limits in accordance with GL 95-05. The definition of DOSE EQUIVALENT I-131 would be replaced with the Improved Standard TS definition wording in the first sentence and an equation would be added based on dose conversion factors derived from the International Commission on Radiation Protection (ICRP) ICRP-30. TS 3.4.8, Specific Activity, would be revised by reducing the DOSE EQUIVALENT I-131 limit from 1.0 μCi/gram to 0.35 μCi/gram for the 48-hour limit and from 60 μCi/gram to 21 μCi/gram for the maximum instantaneous limit. Item 4.a in TS Table 4.4-12, Primary Coolant Specific Activity Sample and Analysis Program, TS Figure 3.4-1, and the Bases for TS 3/4.4.8 are also modified to reflect the reduced DOSE EQUIVALENT I-131 limits. The February 23, 1999, letter providing a revised control room dose calculation; and the June 28, and July 15, 1999, letters providing final copies of the revised TS pages, did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the March 25, 1998, Federal Register notice.

The February 23, 1999, letter provided a revised control room dose calculation in support of both of the amendment requests discussed above. Importantly, this calculation assumed the lower allowable primary-to-secondary leak rate limits associated with the June 18, 1996, submittal, and the reduced RCS specific activity limits associated with the October 22, 1997, submittal. Because of this interdependence, the changes of the first amendment request must be implemented concurrently with those of the second in order for the supporting analysis to remain valid. Hence, both of these license amendment requests have been combined into this single amendment.

2.0 BACKGROUND

The acceptance criteria (i.e., plugging limits) for degraded steam generator tubes are specified in the plant TSs. The traditional strategy for achieving adequate structural and leakage integrity of the degraded tubes has been to establish a minimum wall thickness requirement in accordance with NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR [Pressurized Water Reactor] Steam Generator Tubes." The minimum wall thickness requirement was developed with the assumption of a uniform thinning of the tube wall. This assumed degradation mechanism is inherently conservative for certain forms of tube degradation. Conservative repair limits may lead to removing degraded tubes from service that may otherwise have adequate structural and leakage integrity for further service.

To reduce unnecessary conservatism in the minimum wall thickness requirement for certain degradation, the industry proposed voltage-based repair criteria for ODSCC confined within the thickness of the TSPs. The NRC staff 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 (59 FR 41520). On August 3, 1995, the NRC staff issued GL 95-05, which took into consideration public comments on the draft GL cited above, domestic operating experience under the voltage-based repair criteria, and additional data made available from European nuclear power plants.

The guidance of GL 95-05 does not set depth-based limits on predominantly axially-oriented ODSCC at TSP locations, rather, it relies on empirically derived correlations between a nondestructive inspection parameter, the bobbin coil voltage, and tube burst pressure and leak rate. Although the total tube integrity margins may be reduced following application of a voltage-based repair criteria, the guidance in GL 95-05 ensures structural and leakage integrity continue to be maintained at acceptable levels consistent with the requirements of 10 CFR Part 50 and the guideline values in 10 CFR Part 100. Since the voltage-based repair criteria do not require a minimum tube wall thickness, there is the possibility for tubes with through-wall cracks to remain in service. Because of the increased likelihood of such flaws, the NRC staff included provisions for augmented steam generator tube inspections and restrictive operational leakage limits.

GL 95-05 specifies, in part, that: (1) the repair criteria are only applicable to predominantly axially oriented ODSCC located within the bounds of the TSPs; (2) licensees should perform an evaluation to confirm that the degraded steam generator tubes will retain adequate structural and leakage integrity from cycle to cycle; (3) licensees should adhere to specific inspection criteria to ensure consistency in methods between inspections; (4) tubes must be periodically removed from the steam generators, examined, and destructively tested to verify the morphology of the degradation and provide burst and leakage data for structural and leakage integrity evaluations; (5) the operational leakage limit in the plant TSs should be reduced; (6) licensees should implement an operational leakage monitoring program; and (7) specific reporting requirements should be incorporated into the plant TSs.

Each BVPS unit has three Westinghouse steam generators, which use mill-annealed alloy 600 tubing. These steam generators use steel drilled-hole TSPs and have flow distribution baffle plates. The outside diameter and nominal wall thickness of each tube are seven-eighths of an inch and five-hundredths of an inch, respectively.

3.0 EVALUATION

The licensee stated that it will comply with the guidance in GL 95-05 when implementing its voltage-based alternate repair criteria for BVPS-2 steam generator tubing. However, as permitted by GL 95-05, the licensee proposed to use an industry-proposed and NRC-accepted method to address new bobbin coil probe variability addressed in Section 3.c.2 of Attachment 1 to GL 95-05. The licensee proposed to use alternative methods to Section 3.c.3 of Attachment 1 to GL 95-05 to address probe wear. In addition, the licensee proposed to incorporate verbatim the model TSs in GL 95-05 into the BVPS-2 TSs. The major issues related to the licensee's implementation of the alternate repair criteria are discussed below.

3.1 Tube Repair Limits

The proposed repair criteria will (1) permit degraded tubes having indications confined to within the thickness of the TSPs with bobbin voltages less than or equal to 2.0 volts to remain in service; (2) permit degraded tubes having indications confined to within the thickness of the TSPs with bobbin voltages greater than 2.0 volts but less than or equal to the upper voltage limit to remain in service if inspections using rotating pancake coil probe (or acceptable alternatives) do not detect degradation; and (3) require degraded tubes having indications

confined to within the thickness of the TSPs with bobbin voltages greater than the upper voltage limit be plugged or repaired.

The lower voltage limit of 2.0 volts to be followed by the licensee is consistent with the recommended value specified in GL 95-05 for seven-eighths of an inch of steam generator tubing. The upper voltage limit is derived based on the lower 95 percent prediction interval of the burst pressure versus bobbin voltage correlation, adjusted for lower bound material properties evaluated at the 95 percent confidence level. The upper voltage limit is further reduced to account for uncertainty in the nondestructive examination technique and flaw growth over the next operating cycle. The industry periodically updates the database for burst pressure and bobbin voltage when the destructive test data from pulled tubes are available; therefore, the upper voltage limit may vary as additional data are incorporated into the database.

3.2 Inspection Issues

With respect to probe variability, the licensee proposed to follow an approach developed through the Nuclear Energy Institute (NEI). The proposed procedures and methodology are described in the October 15, 1996, letter from Alex Marion (NEI) to Brian Sheron (NRC). Based on a review of data used originally to support the position that only the primary frequency was required for tests on new probes to verify that they met the voltage variability specification of ±10 percent of the nominal response, the industry indicated that testing at only the primary frequency was not sufficient. The proposed approach specifies that the voltage responses from the primary frequency and mix frequency channels of new probes be within ±10 percent of the nominal voltage responses when voltages are normalized to the 20 percent through-wall flaw values. The nominal voltage responses were established as the average voltages obtained from the American Society of Mechanical Engineers' standard drilled hole flaws for at least 10 production probes. In a letter from Brian Sheron (NRC) to David Modeen (NEI) dated July 29, 1997, the NRC indicated that, based on the NRC staff's review of probe test data, this alternative method to Section 3.c.2 of Attachment 1 of GL 95-05 to address probe variability is acceptable. Therefore, the licensee's proposal to follow the industry approach is acceptable.

Section 3.c.3 of Attachment 1 to GL 95-05 specifies guidance for probe wear. The licensee proposed to use an alternative to Section 3.c.3. The alternative approach, developed through the NEI, specifies that if the probe does not satisfy the voltage variability criterion for wear of ± 15 percent limit before its replacement, all tubes which exhibited flaw signals with voltage responses measured at 75 percent or greater of the lower repair limit must be reinspected with a bobbin probe satisfying the ± 15 percent wear standard criterion. The voltages from the reinspection should be used as the basis for tube repair. The NRC staff completed a review of the NEI proposed alternative method which was provided to the NRC in letters dated February 3, 1995, January 23, and February 23, 1996. The February 23, 1996, letter contained the NEI's agreement to incorporate the NRC staff's recommended changes to the proposed alternative method, which had been provided to the NEI by letter dated February 9, 1996. The NRC staff has concluded that, based on their review of test results for the probes and the NEI's agreement to make recommended changes, the alternative approach is acceptable. This acceptance is documented in a letter from Brian Sheron (NRC) to Alex Marion (NEI) dated March 18, 1996. Therefore, the licensee's proposal to follow the industry approach to address probe wear is acceptable.

3.3 Structural and Leakage Integrity Assessments

GL 95-05 guidance for the voltage-based repair criteria focuses on maintaining tube 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. RG 1.121 recommends that a margin of safety of 1.43 against tube failure under postulated accident conditions and a margin of safety of 3 against burst during normal operation be maintained for steam generator tubes. Because GL 95-05 addresses tubes affected with ODSCC confined to within the thickness of the TSP during normal operation, the NRC staff concluded that the structural constraint provided by the TSP ensures all tubes to which the voltage-based criteria apply will retain a margin of safety of 3 with respect to burst under normal operating conditions. For a postulated main steam line break (MSLB) accident, however, the TSP may displace axially during steam generator blowdown such that the ODSCC affected portion of the tubing may no longer be fully constrained by the TSP. 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.

In order to confirm the structural and leakage integrity of the tube until the next scheduled inspection, GL 95-05 specifies a methodology to determine the conditional burst probability and the total primary-to-secondary leak rate from an affected steam generator during a postulated MSLB event. To complete GL 95-05 prescribed assessments, the licensee proposes to follow the methodology described in WCAP-14277, Revision 1, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," December 1996. The NRC staff has previously reviewed the methodology in WCAP-14277, Revision 1, in a safety evaluation for Joseph M. Farley Nuclear Plant, Unit 2, Docket No. 50-364, dated April 7, 1995, and determined that it is adequate. Additional details relevant to the licensee's application of this methodology to BVPS-2 are provided below.

By a letter dated February 13, 1997, the licensee requested staff approval to use a voltage dependent probability of detection (POD) instead of the constant POD of 0.6 specified by GL 95-05. The voltage dependent POD approach calculates the distribution of bobbin indications as a function of voltage at the beginning of the cycle. The NRC staff is currently addressing such an approach generically through the NEI. Until the work with the NEI is completed and a revised POD is approved by the NRC, DLC must use a constant value of 0.6.

3.3.1 Conditional Probability of Burst

The licensee will use the methodology described in Revision 1 of WCAP-14277 for performing a probabilistic analysis to quantify the potential for steam generator tube ruptures given an MSLB event. The results of the probabilistic analysis will be compared to a threshold value of 1×10^{-2} per cycle in accordance with GL 95-05. This threshold value provides assurance that the probability of burst is acceptable considering the assumptions of the calculation and the results of the NRC 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 TSP could contribute a significant fraction to the overall conditional probability of tube rupture from all forms of degradation assumed and evaluated as acceptable in NUREG-0844. The NRC staff concludes that the licensee's

proposed methodology for calculating the conditional burst probability is consistent with the guidance in GL 95-05 and is acceptable.

3.3.2 Accident Leakage

The licensee will use the methodology described in Revision 1 of WCAP-14277 for calculating the steam generator tube leakage from the faulted steam generator during a postulated MSLB event. The model consists of two major components: (1) a model predicting the probability that a given indication will leak as a function of voltage (i.e., the probability of leakage model); and (2) a model predicting leak rate as a function of voltage, given that leakage occurs (i.e., the conditional leak rate model). The NRC staff concludes that the licensee's proposed methodology for calculating the tube leakage is consistent with the guidance in GL 95-05 and is acceptable.

3.3.3 Primary-to-Secondary Leakage

Because the voltage-based repair criteria would allow degraded tubes to remain in service, the degraded tubes may develop through-wall cracks which may leak during normal operation, transients, or postulated accidents. Therefore, as a defense-in-depth measure, GL 95-05 specifies that the operational leakage limits of the plant TSs be limited to 150 gpd from any one steam generator. The licensee proposed to change the leakage limits in the plant TSs to 150 gpd through any one steam generator. In addition, the licensee has incorporated the guidelines in the Electric Power Research Institute's Report, TR-104788, "PWR Primary-to-Secondary Leak Guidelines," into the BVPS-2 operating procedures. The NRC staff concludes that the proposed operational leakage limit of 150 gpd through any one steam generator for BVPS-2 TSs is consistent with GL 95-05 and, therefore, is acceptable.

3.4 Degradation Monitoring

Voltage-based steam generator tube repair criteria are applicable only to axially-oriented ODSCC confined to the TSP intersections. To confirm the nature of the degradation at these locations, tubes are periodically removed from the steam generators for destructive tests. The test data from removed tubes can confirm that the nature of the degradation observed at these locations is predominantly axially-oriented ODSCC, provide data for assessing the reliability of the inspection methods, and supplement the existing databases (e.g., burst pressure, probability of leakage, and leak rate). GL 95-05 specifies that at least two tubes be removed from steam generators with the objective of retrieving as many intersections as practical (minimum of four intersections) either during the plant steam generator inspection outage that implements the voltage-based repair criteria or during an inspection outage preceding initial application of these criteria. On an ongoing bases, additional tube specimens (minimum of two intersections) should be removed at the first refueling outage following 34 effective full power months of operation or at the maximum interval of three refueling outages after the previous tube pull. Alternatively, the licensee may participate in an industry-sponsored tube pull program endorsed by the NRC staff as described in GL 95-05. The licensee plans to implement a tube pull schedule in accordance with GL 95-05 to verify the morphology at the TSPs. It should be noted that the results of pulled-tube examinations will have to confirm the morphology of the cracks as being axially-oriented ODSCC. The alternate tube plugging criteria do not apply to cracks that are determined to be anything other than axially-oriented ODSCC.

3.5 Evaluation of Exclusion of Tubes from Alternate Plugging Criteria Due to Combined Loss-of-Coolant-Accident plus Safe Shutdown Earthquake Loadings

The combined effects of Loss-of-Coolant-Accident (LOCA) plus Safe Shutdown Earthquake (SSE) loads as they relate to the application of Alternate Plugging Criteria (APC) have been addressed by the licensee in its submittal of June 18, 1996. Westinghouse has determined that tube collapse may occur in the steam generators at some plants under combined LOCA + SSE loadings. Lateral loads at the wedge supports on the periphery of the plates may initially deform the tubes in the vicinity of the wedge supports during a postulated LOCA accident in combination with an SSE. In addition, the pressure differential existing across the tubes during a LOCA 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 RCS 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. Second, there is a potential that partial through-wall cracks in tubes could progress to complete through-wall cracks during tube deformation or collapse.

In developing an APC, where tubes with specified types of degradation at the TSP could remain in service beyond normal degradation levels, it is necessary to exclude tubes from the APC that experience permanent deformation under LOCA plus SSE. Permanent tube deformation for this evaluation is defined as a change in diameter that exceeds 0.025 inch.

A similar analysis was performed previously for BVPS-1. However, a separate analysis was necessary for BVPS-2, as the BVPS-2 steam generators are Series 51M design, as opposed to Series 51 for BVPS-1. There are several differences between the two generator designs that affect the analysis in question. The most fundamental difference between the Series 51 and 51M designs insofar as this analysis is concerned is the number of wedge groups. For the Series 51 design, there are six wedge groups located uniformly around the plate circumference. For the Series 51M, the number of wedge groups is increased to 12 for TSP 1, and to 14 for the remaining plates. There is also a flow distribution baffle (FDB) added to the Series 51M, giving a total of eight TSP, as opposed to seven for the Series 51. The FDB has been added to the design between the old TSP 1 position and the tubesheet. The axial position of the remaining Series 51M plates is identical to the Series 51 plates.

In calculating the wedge loads, it was assumed that the in-plane loads are reacted by the various wedge groups based on a cosine distribution of the load. For the LOCA rarefaction load, which acts in the plane of the U-bend, the load is treated as uni-directional. For the seismic load, however, the load can act randomly, such that the direction of the load can act normal to any given wedge location.

The number of deformed tubes at each wedge location is calculated based on some plate crush tests performed for the Series 51 plates. While none of the samples tested are directly applicable to the Series 51M wedge location and hole pattern, sufficient information is available from the test data to draw conclusions related to the Series 51M plates of the BVPS-2 steam generators. The closest samples are the two test pieces with the load applied normal to the hole array. Although the orientation of the holes relative to the load direction is not prototypic, the hole pattern for these samples (hole diameters and hole pitch), are prototypic for the Series

51M plates. The test results are, therefore, applicable to Series 51M plates of the BVPS-2 steam generators.

The test results for the number of deformed tubes as a function of load for the most applicable hole arrangement show that for an elastic load less than the threshold load for tube deformation, there will be no deformed tubes. The test results show a load threshold for tube deformation that is several times higher than the maximum wedge loads. Accordingly, the maximum wedge load for the BVPS-2 plates is well below the threshold for tube deformation.

3.6 Assessment of Radiological Consequences

The licensee proposes to change their TSs to implement steam generator tube alternate repair criteria per the requirements of NRC GL 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking." The licensee has requested that the specific activity limits of dose equivalent iodine ^{131}I in the primary coolant be established at 0.35 μCi/gram for the 48-hour limit and at 21μCi/gram for the maximum instantaneous limit (in accordance with GL 95-05). The allowable activity level of dose equivalent ^{131}I in the secondary coolant was assumed to be equal to the TS limit of 0.1μCi/gram. The licensee also requested that BVPS-2 be approved to operate based upon a 3.0 gpm primary-to-secondary leak initiated by an accident in the faulted steam generator in addition to the TS allowable value for primary-to-secondary leakage from each of the two intact steam generators of 150 gpd per steam generator. As part of these amendment requests, the licensee performed an assessment of the radiological dose consequences of an MSLB accident. The licensee found the radiological consequences of incorporating these proposed changes to be acceptable based on the NRC acceptance criteria doses at the Exclusion Area Boundary (EAB), the Low-Population Zone (LPZ), and the control room.

The NRC staff reviewed the licensee's calculations and performed confirmatory calculations to check the acceptability of the licensee's methodology and resulting doses. As part of the NRC staff's review, the doses resulting from an MSLB accident were calculated using the methodology associated with the Standard Review Plan (SRP), Section 15.1.5, Appendix A (NUREG-0800). The NRC staff performed two separate assessments: one based upon a pre-existing iodine spike activity level of 21μCi/gram of dose equivalent ^{131}I; the other based upon an accident initiated iodine spike. For the accident initiated spike, the staff assumed that the primary coolant activity level was 0.35 μCi/gram of dose equivalent ^{131}I. The accident initiated an increase in the release rate of iodine from the fuel by a factor of 500 over the release rate to maintain an activity level of 0.35 μCi/gram of dose equivalent ^{131}I in the primary coolant. The licensee had assumed that the accident initiated (co-incident) iodine spike ceases after 4 hours. The staff calculated that the iodine in the fuel gap would not be exhausted prior to 8 hours. Therefore, the NRC staff performed its dose assessment for the accident initiated spike case using the assumption that the accident initiated spike lasts for the full 8-hour duration of an MSLB accident. For these two cases (pre-existing and accident initiated iodine spikes), the staff calculated doses for individuals located at the EAB and at the LPZ. The NRC staff also calculated the thyroid dose to the control room operators. The parameters which were utilized in the NRC staff's assessment are presented in Table 1, which is attached. The value used for the efficiency of the control room intake filter was slightly reduced (below the licensee's number) to account for one percent bypass flow. The EAB, LPZ, and control room doses calculated by the NRC staff are presented in Table 2, which is attached.

The NRC staff's calculations confirmed that the doses from a postulated MSLB accident meet the acceptance criteria set forth in the SRP, and that the licensee's calculations are acceptable. The results of both the licensee's and the NRC staff's calculations showed that the thyroid doses at the EAB and LPZ would be less than the guidelines established by SRP 15.1.5, Appendix A, which implement the relevant requirements of 10 CFR Part 100. Those acceptance criteria are: 300 rem thyroid dose at the EAB and LPZ for the pre-existing spike case, and 30 rem thyroid dose at the EAB and LPZ for the accident initiated spike case. Additionally, the control room operator thyroid dose would be less than the guideline acceptance criteria of 30 rem thyroid dose to the control room operators which are established in SRP 6.4. The SRP 6.4 guidelines implement the relevant requirements of 10 CFR 50, Appendix A, General Design Criterion 19. On this basis, the NRC staff finds the radiological consequences of the licensee's request to implement steam generator tube alternate repair criteria in accordance with GL 95-05 at BVPS-2 to be acceptable. Use of this steam generator alternate repair criteria will permit the licensee to maintain specific activity limits of dose equivalent ^{131}I in the primary coolant of 0.35 μCi/gram for the 48-hour limit and at 21μCi/gram for the maximum instantaneous limit, as well as restricting the allowable maximum primary-to-secondary coolant leakage to 3.1 gpm (Note: this 3.1 gpm is composed of the 3.0 gpm leakage initiated by the MSLB event and the .1 gpm contribution from the 150 gpd leakage assumed to exist at the time of the MSLB).

3.7 Summary

As set forth above, the NRC staff has reviewed the proposed amendments to implement the voltage-based repair criteria for steam generator tubes and implementation of reduced RCS specific activity limits at BVPS-2 in accordance with GL 95-05. The NRC staff concludes that adequate structural and leakage integrity of steam generator tubing can be assured, consistent with applicable regulatory requirements, for indications to which the voltage-based repair criteria will be applied. The GL 95-05 guidelines specify successful demonstration for each inspection outage that the conditional probability of burst and the primary-to-secondary leakage during a postulated MSLB will be acceptable. The NRC staff finds that the proposed alternate repair criteria and the reduced RCS specific activity limits are consistent with GL 95-05 and thus are acceptable. The licensee may incorporate the proposed changes into the BVPS-2 TSs.

Based on the margin identified in the submittal between the maximum wedge load for the Series 51M plates and the threshold for permanent tube deformation in the plate crush tests, the NRC staff concurs with the licensee's determination that there will be no tubes which are likely to experience permanent diametral deformation greater than twenty-five-hundredths of an inch due to combined LOCA plus SSE. Thus, it is not necessary to exclude any tube from the APC criteria for reasons of tube deformation for BVPS-2.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements 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 proposed findings that the amendment involves no significant hazards consideration, and there has been no public comment on such findings (63 FR 64109 and 63 FR 14485). 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.

6.0 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.

Attachments: 1.) Input Parameters for BVPS-2 Evaluation of MSLB Accident
2.) Thyroid Doses from BVPS-2 MSLB Accident (REM) Values Calculated by NRC Staff)

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TABLE 1
INPUT PARAMETERS FOR BEAVER VALLEY POWER STATION, NO. 2
EVALUATION OF MAIN STEAM LINE BREAK ACCIDENT

1. Primary Coolant Concentration of 21 μCi/g of Dose Equivalent ^{131}I

	Pre-existing Spike Value (μCi/g)
^{131}I =	16.46
^{132}I =	6.51
^{133}I =	24.62
^{134}I =	3.62
^{135}I =	14.25

2. Data on Primary Coolant and Secondary Coolant

Primary Coolant Volume (ft^3)	7,791
Excluding Pressurizer volume	7,451
Primary Coolant Temperature (°F)	576.6
Secondary Coolant Steam Mass (+10%)(lb/SG)	6,630
Secondary Coolant Liquid Mass (+10%)(lb/SG)	162,946
Secondary Coolant Steam Temperature (°F)	600

3. TS Limits for DE ^{131}I in the Primary and Secondary Coolant

Maximum Instantaneous DE ^{131}I Concentration (μCi/g)	21
Primary Coolant DE ^{131}I Concentration (μCi/g)	0.35
Secondary Coolant DE ^{131}I Concentration (μCi/g)	0.1

4. TS Value for the Primary to Secondary Leak Rate

Primary to secondary leak rate, maximum any SG (gpd)	150
Primary to secondary leak rate, total all SGs (gpd)	450

5. Maximum Primary to Secondary Leak Rate to the Faulted and Intact SGs

Faulted SG (gpm)	3.1
Intact SGs (gpm/SG)	0.1

6. Iodine Partition Factor

Faulted SG		1.0
Intact SG:	initially	1.0
	after one hour	0.01

7. Steam Released to the Environment

Faulted SG (0-0.5 hours)	169,306 lbs
(0.5-8 hours)	11,652 lbs
Intact SGs (0 - 2 hours)	336,776 lbs
(2 - 8 hours)	705,393 lbs

8. Letdown Flow Rate (gpm) 120

9. Release Rate for 0.35 μCi/g of Dose Equivalent ^{131}I

	Release Rate (Ci/hr)	500X Release Rate (Ci/hr)
^{131}I =	5.34	2670
^{132}I =	6.93	3460
^{133}I =	9.94	4970
^{134}I =	8.23	4110
^{135}I =	8.27	4140

10. Atmospheric Dispersion Factors

	sec/m^3
EAB (0-2 hours)	1.25×10^{-3}
LPZ (0-8 hours)	6.04×10^{-5}
Control Room (0-8 hours)	1.59×10^{-4}

11. Control Room Parameters

Filter Efficiency (%)	
Air intake filter	95 (effective 94.05)
Volume (ft^3)	1.73×10^5

	Time after accident (hrs)				
	0-.5	.5-1.5	1.5-8	8-8.5	>8.5
Makeup flow (cfm)	500	0	1,030	16,900	500
Recirculation Flow (cfm)	0	0	0	0	0
Unfiltered Inleakage (cfm)	0	10	10	0	0

Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
4-30 days	0.4

Table 2
THYROID DOSES FROM BEAVER VALLEY POWER STATION, NO.2
MAIN STEAM LINE BREAK ACCIDENT (REM)
(VALUES CALCULATED BY NRC STAFF)

LOCATION	DOSE	
	Pre-Existing Spike	Accident-Initiated Spike**
EAB	13.14*	16.76
LPZ	2.03*	8.70
Control Room **	0.55	0.77

* Acceptance Criterion = 300 rem to the thyroid
** Acceptance Criterion = 30 rem to the thyroid