From:	Lashley, Phil H
To:	Tobin, Jennifer
Subject:	[External_Sender] Beaver Valley Power Station, Units 1 and 2 - Issuance of Amendment Nos. 305 and 195 to Modify Primary and Secondary Coolant Activity Technical Specifications (EPID L-2019-LLA-0223)
Date:	Wednesday, September 30, 2020 5:28:44 PM
Attachments:	Comments on the Safety Evaluation for BVPS Amendments 305 and 195.pdf CR 2020-07603.pdf

Jenny,

As we discussed on the phone, attached are the comments that we noted on the Safety Evaluation. Also attached is a condition report generated to document the erroneous number in the Westinghouse calculation.

Respectfully,

Phil H. Lashley
Manager, Fleet Licensing
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Figure 3.4.16-1 would be annotated with a "(Unit 1)" at the end of the title, and a new unit-specific figure for reactor coolant DE I-131 specific activity limit versus percent of rated thermal power would be added as Figure 3.4.16-2.

The LCO statement for TS 3.7.13 would be changed as indicated below by the addition of the **bold, underlined text**:

LCO 3.7.13 The specific activity of the secondary coolant shall be $\leq$ 0.10 μCi/gm DOSE EQUIVALENT I-131 **(Unit 1), and $\leq$ 0.05 μCi/gm DOSE EQUIVALENT I-131 (Unit 2).**

SR 3.7.13.1 would be changed as indicated below by the addition of the **bold, underlined text**:

SR 3.7.13.1 Verify the specific activity of the secondary coolant is $\leq$ 0.10 μCi/gm DOSE EQUIVALENT I-131 **(Unit 1), and $\leq$ 0.05 μCi/gm DOSE EQUIVALENT I-131 (Unit 2).**

The proposed revisions to TS 5.5.7 for the CREVS change the acceptance criteria for the CREVS penetration and system bypass requirement and CREVS charcoal adsorber removal efficiency from "99%" to "99.5%" for both units based on the licensee's evaluation in the license amendment request (LAR).

This sentence could imply that penetration and system bypass is changing from 99% to 99.5%. The penetration and bypass change is from < 0.05% to < 0.5%.

The proposed change to the C
Controls Section of the TSs w
unfiltered air inleakage test fre
The note would modify the requirements listed in TS 5.5.14.c for determining the unfiltered air inleakage past the CRE boundary into the CRE and would state:

- NOTE -

The three-year test frequency for the CRE unfiltered air inleakage test failure that occurred in October 2017 may be extended an additional three years, not to exceed October 2023.

3.0 TECHNICAL EVALUATION

In Sections 3.1 and 3.2 below, the NRC staff describes the evaluations and analyses provided by the licensee to justify the proposed changes to TSs 3.4.16, 3.7.13, 5.5.7, and 5.5.14.c.

3.1 Radiological Consequences of Design-Basis Accidents

Background

The licensee has implemented alternate source term (AST) methodology in the current licensing bases (CLB) at Beaver Valley, Units 1 and 2, through various phases. The licensee first requested a selective implementation of the AST for the fuel handling accident (FHA) by letter dated March 19, 2001 (ADAMS Accession No. ML010810433). The NRC staff approved the selective AST evaluation of the FHA radiological consequence analysis with License Amendment Nos. 241 and 121 for Units 1 and 2, respectively, dated August 30, 2001 (ADAMS Accession

The licensee's dose calculation model used the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent from external exposure from submersion due to halogens and noble gases transported to offsite locations (EAB and LPZ) and in the CR. The CEDE is calculated using the dose conversion factors (DCFs) from Environmental Protection Agency (EPA)-520/1-88-020, Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," dated September 1988, which uses the methodology provided in the International Commission on Radiological Protection, ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers," dated 1979-1988. As described in Sections 4.1.4 and 4.2.7 of RG 1.183, the effective dose equivalent may be used in lieu of deep dose equivalent in determining the contribution of external dose to the TEDE if the whole body is irradiated uniformly. The effective dose equivalent is calculated using the DCFs from EPA-402-R-93-081, Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," dated September 1993, to determine the TEDE dose as is required for AST evaluations. To calculate the direct shine dose to the CR operator, the licensee used the DCFs from American National Standards Institute/American Nuclear Society, ANSI/ANS 6.1.1-1991, "Neutron and Gamma-Ray Fluence-to-Dose Factors," dated 1991. The use of this dose methodology and DCFs is consistent with the guidance in RG 1.183 and is, therefore, acceptable to the NRC staff.

The results of the evaluations performed by the licensee, as well as the applicable dose acceptance criteria from RG 1.183, are shown in Table 1 in Section 5 of this SE. CR atmospheric dispersion factors are shown in Tables 2A through 2H in Section 5 of this SE. Offsite atmospheric dispersion factors are shown in Table 3 and CR data and assumptions are shown in Table 4 in Section 5 of this SE. Accident-specific data and assumptions are shown in Tables 5 through 11 in Section 5 of this SE.

3.1.1 Loss-of-Coolant Accident

The radiological consequence design-basis LOCA analysis is a deterministic evaluation based on the assumption of a major rupture of the primary RCS piping. The accident scenario assumes the deterministic failure of the emergency core cooling system (ECCS) to provide adequate core cooling, which results in a significant amount of core damage as specified in RG 1.183. This general scenario does not represent any specific accident sequence but is representative of a class of severe damage incidents that was evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the severity of incidents evaluated for design-basis transient analyses.

The LOCA considered in this evaluation is a complete and instantaneous circumferential severance of the primary RCS piping, which would result in the maximum fuel temperature and primary containment pressure among the full range of LOCAs. Due to the postulated loss of core cooling, the fuel heats up, resulting in the release of fission products. The fission product release is assumed to occur in phases over a 2-hour period.

When using the AST for the evaluation of a design-basis LOCA for a PWR, it is assumed that the initial fission product release to the containment will last for 30 seconds and will consist of the radioactive materials dissolved or suspended in the RCS liquid. After 30 seconds, fuel damage is assumed to begin and is characterized by clad damage that releases the fission product inventory assumed to reside in the fuel gap. The fuel gap release phase is assumed to continue until 30 minutes after the initial breach of the RCS. As core damage continues, the

The duration of the fuel gap release phase is 30 minutes. The phase lasts until 30.5 minutes after the initial breach of RCS. Ref. UR(B)-487. Rev. 3, p. 23.

This wording implies the containment is mixed once per 2 hours. The mixing rate is 2 unsprayed volumes per hour. Ref. UR(B)-487 R3, p. 24

The licensee stated that the containment atmosphere is mixed following a LOCA by four mechanisms: 1) momentum transfer from the fluid jet exiting the break, 2) momentum transfer from the spray droplets to the surrounding gas, 3) forced and natural convection flows within the containment atmosphere, and 4) molecular diffusion. The licensee stated that these mechanisms work together to enhance mixing within the containment to provide a homogeneous gas mixture and prevent local accumulation of fission products. Therefore, in accordance with the current CLB, the mixing rate between the effectively sprayed volume and unsprayed volume of the containment is assumed to be 2 hours (hr), which is the default rate permitted by NUREG-0800 Section 6.5.2. The methodology used by the licensee to determine the containment mixing rate was previously approved by the NRC in License Amendment No. 280 for Unit 1 and License Amendment No. 164 for Unit 2. Therefore, the NRC staff continues to find the licensee's methodology acceptable.

3.1.1.4 Emergency Core Cooling System Leakage Outside of the Containment

During a DBA LOCA, some fission products released from the fuel will be carried to the containment sump via spillage from the RCS or by transport of activity from the containment atmosphere to the sump by containment sprays and natural processes, such as deposition and plateout. During the initial phases of a LOCA, safety injection (SI) and containment spray systems draw water from the RWST. The licensee assumed that at 1,200 seconds (5 minutes) after the start of the LOCA event, these systems would start to draw water from the containment sump instead of the RWST. This recirculation flow causes contaminated water to be circulated through piping and components outside of the containment where small amounts of system leakage could provide a path for the release of fission products to the environment.

1,200 seconds = 20 minutes.
1,200 seconds is the correct input
(ESF leak duration = 1200 sec to 30 days, Ref. UR(B)-487 R3)

The licensee assumed that all t[illegible] moved to the containment sump [illegible] atmosphere. The remaining rad[illegible] particulates that will not become [illegible] assumption is conservative in that all the radioiodines released from the fuel are assumed to reside in both the containment atmosphere and containment sump. The licensee assumed that the leakage rate is two times the expected value of 5,700 cubic centimeters (cc)/hr (the bounding value for Unit 1; the Unit 2 value is 2,134 cc/hr) or 11,400 cc/hr. The licensee assumed that 10 percent of the entrained iodine activity is released to the atmosphere of the surrounding auxiliary building. The licensee assumed that this activity is exhausted without holdup, mixing, or filtration, and that the chemical form of the iodine released is 97 percent elemental and 3 percent organic. Because the licensee used data and assumptions consistent with the guidance in RG 1.183 and the CLB, the NRC staff finds the licensee's assessment of the dose consequence from ECCS leakage to be acceptable.

3.1.1.5 Release from Reactor Water Storage Tank Due to Emergency Core Cooling Back-Leakage

Although the RWST is isolated during recirculation, design leakage through ECCS valving provides a pathway for back-leakage of the containment sump water to the RWST. The RWST is located in the plant yard and is vented to the atmosphere. Since this release path represents a bypass of the containment, dose consequences must be considered. The concentration of radionuclides in the containment sump water is as modeled above for ECCS leakage. The licensee assumed that containment sump water leaks into the RWST at a rate of 2 gallons per minute (gpm), which is twice the surveillance limit of 1 gpm. The licensee assumed that the leakage into the RWST begins at about 30 minutes (1,768 seconds) post-LOCA. The licensee assumed that radioiodines are projected to be released via the RWST vent starting at about 50 minutes (3,039 seconds) post-LOCA and continues until the end of the accident analysis period of 30 days.

The licensee assumed that a portion of the iodine dissolved in the back-leakage will be retained within the RWST. The time-dependent iodine release rates used by the licensee are illustrated in Figure 7.2-3 and Table 7.2-3 of the technical report dated August 12, 2019, and were included in the April 14, 2020, supplement to this LAR. Values range from about 0.01 per day at 3,039 seconds post-accident to a minimum value of about 2.0 x 10^{-7} per day at 720 hours. The licensee assumed that this activity is exhausted without filtration and that the chemical form of the iodine released is 97 percent elemental and 3 percent organic. The methodology used by the licensee to determine the RWST release fractions was previously approved by the NRC in License Amendment No. 280 for Unit 1 and for License Amendment No. 164 for Unit 2.

3.1.1.6 Control Room Occupancy Dose

The licensee's assumptions used for the evaluation of CR habitability are discussed in Section 3.1.9 of this SE and presented in Table 4 in Section 5 of this SE. Provided below are the critical LOCA-specific assumptions associated with CR response and activity transport.

- In accordance with the Beaver Valley CLB, due to the rapid pressure transient expected following a LOCA, the Containment Isolation Phase B (CIB) signal, which initiates the CR isolation and emergency ventilation following a LOCA is assumed to actuate at t = 0 hours.

- Considering a LOOP, the maximum estimated delay in attaining CR isolation after receipt of a CIB signal to switch CR ventilation from normal mode to emergency mode is 77 seconds, which accounts for delays due to emergency diesel generator (EDG) start and load sequencing (including time for damper movement/realignment).

- No credit is taken for automatic initiation of the Unit 2 CREVS; rather, it is assumed the CREVS will be initiated by manual operator action providing a pressurized CR within 30 minutes of accident initiation.

This should be "isolation" mode. "Emergency" mode is often synonymous with Pressurization Mode and CREVS operation.

The licensee evaluated the following LOCA sou[rces] operator dose due to direct shine:

- direct shine from the containment structure

- direct shine from the airborne source located in the cable spreading room below the Unit 2 portion of the combined CR through floor penetrations

- direct shine from the airborne source located in the cable tray mezzanine area below the Unit 1 portion of the combined CR through floor penetrations

- direct shine through the CR walls and floor from the CREVS intake filters located in the adjacent room (Unit 2 filters) and below (Unit 1 filters), respectively break flow into the ruptured SG flashes and is released to the condenser for a short duration prior to the reactor trip, and thereafter, directly to the environment via the MSSVs/ADVs. The remaining break flow mixes with the secondary side liquid and is released to the environment via steam releases through MSSVs/ADVs. The activity in the RCS also leaks into the intact SGs via SG tube leakage and is released to the environment from the MSSVs/ADVs.

The NRC staff previously evaluated the radiological consequences for the MSLB and SGTR accidents in License Amendment No. 273 for Unit 1 in support of Unit 1 operation with the RSGs at the anticipated EPU reactor core power level of 2,918 MWt and implementing the AST. The NRC staff previously evaluated the radiological consequences for the MSLB and SGTR accidents for Unit 2 in License Amendment Nos. 275 and 156 in support of the Beaver Valley EPU.

Appendix F of RG 1.183 identifies acceptable radiological analysis assumptions for an SGTR. The licensee's analysis indicates that no fuel melting or fuel cladding failure is postulated for the SGTR event. Therefore, in accordance with RG 1.183, the licensee assumed that the activity released is based on the maximum coolant activity allowed by the Unit 1 TSs. The staff notes that the licensee used the higher Unit 1 TS values of allowable coolant activity for both the Unit 1 and 2 SGTR assessments. Therefore, the Unit 2 SGTR analysis has added conservatism since the lower coolant values, as governed by the proposed Unit 2 TSs, are not credited in the analysis. Following the guidance in RG 1.183, the licensee considered two spiking scenarios – a pre-accident iodine spike, which reflects the maximum allowable Unit 1 TS iodine spike activity level, and an accident-initiated iodine spike, which results in an increase in the iodine appearance rate from the fuel to the RCS by a factor of 335.

Primary-to-secondary leakage is assumed to be 150 gpd into the bulk water of the ruptured SG and 300 gpd total into the bulk water of the two intact SGs, as permitted by the Unit 2 TSs. The iodine activity from the flashed portion of the break flow through the ruptured SG is assumed to be directly released to the environment with no iodine partitioning. The radionuclides in the intact SG bulk water are assumed to become vapor at a rate that is a function of the steaming rate for the SGs and the partition coefficient. The licensee assumed that the radionuclide concentration in the SG is partitioned such that 1.0 percent of the radionuclides in the unaffected SG bulk water enter the vapor space and are released to the environment. The steam release from the unaffected SGs continues for approximately 8 hours until the RHR shutdown cooling system can be used to complete the cooldown.

The licensee claimed no credit fo
event and assumed the CR is ma
the environmental release at 8 h
30 minutes.

This applies to iodine only, not all radionuclides.
Ref. UR(B)-219 R3, p. 12
UR(B)-496 R3, p. 11

The NRC staff reviewed the licensee's analysis presented in Calculation No. 8700-UR(B)-219, Revision 3, "Site Boundary and Control Room Doses following a Steam Generator Tube Rupture (SGTR) based on Core Uprate and Alternative Source Term" (Unit 1), and Calculation No. 10080-UR(B)-496, Revision 3, "Site Boundary and Control Room Doses following a Steam Generator Tube Rupture based on Core Uprate and Alternative Source Term Methodology" (Unit 2), provided in the supplement to the LAR dated June 9, 2020. In Revision 3 of Calculation No. 8700-UR(B)-219 and Calculation No. 10080-UR(B)-496 for the postulated SGTR, the licensee considered: a) an increase in allowable unfiltered inleakage into the CRE (inclusive of that associated with ingress /egress) from 30 cfm to 165 cfm and b) review/update (as needed) of all design input parameter values/references to reflect current plant design.

30 to 165 cfm is the change to pressurization mode inleakage. The SGTR does not credit CREVS and pressurization mode is not used. Change to 500 cfm to 1250 cfm unfiltered intake plus inleakage.

consequences are due to leakage of the radioactive reactor primary coolant to the SGs and from there to the environment. Because the LOOP renders the main condenser unavailable, the plant is cooled down by release of steam to the environment through ADVs and MSSVs. The releases to the environment are assumed to continue for 8 hours, at which time shutdown cooling is initiated via operation of the RHR system. Appendix G of RG 1.183 identifies acceptable radiological analysis assumptions for an LRA.

The licensee assumed that due to the LRA rendering the affected RCP inoperable, the resulting loss of primary coolant circulation may result in as much as 20 percent of the core fuel rods experiencing a departure from nucleate boiling. This will cause fuel cladding damage and release of the gap activity in the damaged fuel into the RCS.

In its LAR dated October 20, 2019, the licensee proposed to use the fission product inventory fuel gap fractions from Table 3 of NRC DG-1199 when assessing the dose consequences of Beaver Valley non-LOCA events other than reactivity-initiated accidents where only the fuel clad is postulated to be breached. The licensee stated in its LAR that this change in licensing basis is intended to support flexibility in future Beaver Valley fuel management schemes and is deemed to be acceptable since Beaver Valley falls within, and intends to operate within, the maximum allowable power operating envelope for PWRs shown in Figure 1 of DG-1199. The use of gap fractions from DG-1199 has been previously approved in the NRC staff's SER for the Diablo Canyon Nuclear Power Plant, Units 1 and 2, AST license amendments dated April 17, 2017.

In addition, the licensee proposed to change the radial peaking factor from the CLB value of 1.75 to 1.7. In the technical report dated August 12, 2019, submitted with the April 14, 2020, supplement to the LAR, the licensee states that the reduction in the peaking factor margin to 1.7 continues to be bounded by the radial peaking factor, which is currently designed to 1.62 per the COLRs in Beaver Valley, Unit 1, LRM, Revision 102, and Beaver Valley, Unit 2, LRM, Revision 94.

The radionuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the RCS and transported to the secondary side via primary-to-secondary leakage of 450 gpd for all three SGs for 8 hours. The licensee assumed that this leakage mixes with the bulk water of the SG's secondary side, and that the radionuclides in the bulk water become vapor at a rate that is a function of the steaming rate for the SGs and the partition coefficient. The licensee computed the CR doses for this event assuming that the CR remained in the normal ventilation mode for the duration of the accident.

As described in Sections 3.1.1.6 and 3.1.4 of this SE, the PERC2 code is used for calculating the airborne dose to the CR operator and to a member of the public located at the EAB and LPZ. The NRC staff reviewed the licensee's analysis presented in Calculation No. 10080-UR(B)-493, Revision 1, "Site Boundary and Control Room Doses based on Core Uprate and Alt

unfiltered intake plus inleakage

ving a) a Locked Rotor Accident b) a Loss of AC Po AR dated June 9, 2020. In Revision 1 of Calculation No. 10080-UR(B)-493 for the postulated LRA, the licensee considered: a) an increase in allowable unfiltered inleakage into the CRE (inclusive of that associated with ingress/egress) from 30 cfm to 165 cfm, b) use of fuel gap fractions from Table 3 of DG-1199 for all non-LOCA events that experience fuel damage with the exception of the CREA, and c) review/update (as needed) of all design input parameter values/references to reflect current plant design. The licensee concluded from its analysis that the Beaver Valley site boundary and

CREVS not credited for LRA.
Change to "500 cfm to 1250 cfm"

Percentage of core inventory in melted fuel released to reactor coolant	100% noble gas; 50% halogens
Iodine species released to environment	97% elemental; 3% organic
Iodine partition coefficient	100 (all tubes submerged)
Fraction of noble gas released	1.0 (released without holdup)
Minimum post-accident SG liquid mass	99,217 lbm per SG
Steam releases per SG	
0 to 150 secs	
150 to 300 secs	
300 to 2500 secs	
2500 secs to 8 hrs	
8 hrs to 30 days	
Termination of Release from SGs	
Environmental Release Point	MSSVs/ADVs

Should be 101,799 lbm/SG per Ref. UR(B)-488 R1, p. 13 and L-20-097, Attachment 3.

CREVS Initiation Signal Timing

Emergency mode initiation time	t = 30 minutes (manual initiation)

Table 7A
Beaver Valley, Units 1, Data and Assumptions for the MSLB Accident

Source Term Parameters

Core Power Level	2,918 MWt
Minimum Reactor Coolant Mass	345,097 lbm
Leakrate into Faulted SG	150 gpd at STP
Amount of Accident-Induced Leakage into Faulted SG	N/A
Maximum Time to Cool RCS to 212 ºF	19 hrs
Leakrate into Intact SGs	300 gpd total from 2 SGs at STP
RCS TS Iodine and Noble Gas Activity Concentration	TR[1] Table 4.2-1A (0.35 µCi/gm
Reactor Coolant Equilibrium Iodine Appearance Rates	TR[1] Table 4.2-2A (0.35 µCi/gm
Pre-Accident Iodine Spike Activity	TR[1] Table 4.2-2A (21 µCi/gm
Concurrent Iodine Spike Appearance Rate	500 Xs equilibrium appearance
Duration of Concurrent Iodine Spike	4 hours

Secondary System Release Parameters

Iodine Species released to Environment	97% elemental; 3% organic
Secondary Coolant TS Iodine Activity Concentration	TR[1] Table 4.2-1A (0.10 µCi/gm
Iodine Partition Coefficient in Intact SG	100 (all tubes submerged)
Fraction of Noble Gas Released from Intact SG	1.0 (released without holdup)
Fraction of Iodine Released form Faulted SG	1.0 (released without holdup)
Fraction of Noble Gas Released from Faulted SG	1.0 (released without holdup)
Initial and Minimum Post-Accident Intact SG Liquid Mass	101,799 lbm per SG