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

May 31, 2019

Mr. Adam C. Heflin
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION, UNIT 1 - ISSUANCE OF AMENDMENT NO. 221 RE: TRANSITION TO WESTINGHOUSE CORE DESIGN AND SAFETY ANALYSES INCLUDING ADOPTION OF ALTERNATIVE SOURCE TERM (CAC NO. MF9307; EPID L-2017-LLA-0211)

Dear Mr. Heflin:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 221 to Renewed Facility Operating License No. NPF-42 for the Wolf Creek Generating Station, Unit 1 (Wolf Creek). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 17, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17054C103), as supplemented by letters dated March 22, May 4, July 13, October 18, and November 14, 2017; January 15, January 29, April 19, June 19, August 9, November 15 (two letters), and December 6, 2018; and March 5, May 2, and May 15, 2019 (ADAMS Accession Nos. ML17088A635, ML17130A915, ML17200C939, ML17297A478, ML17325A982, ML18024A477, ML18033B024, ML18114A115, ML18177A198, ML18232A058, ML18325A144, ML18325A145, ML18352A745, ML19070A139, ML19133A057, and ML19148A571, respectively),

The amendment revises Wolf Creek TSs to replace the existing methodology for performing core design, non-loss-of-coolant-accident and loss-of-coolant-accident safety analyses with standard Westinghouse Electric Corporation developed and NRC-approved analysis methodologies. In addition, the proposed amendment revises the Wolf Creek licensing basis by adopting the alternative source term radiological analysis methodology in accordance with Title 10 of the *Code of Federal Regulations* Section 50.67, "Accident source term." This amendment request represents a full scope implementation of the alternative source term as described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

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

Sincerely,

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures:
1. Amendment No. 221 to NPF-42
2. Safety Evaluation

cc: Listserv

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

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION, UNIT 1

DOCKET NO. 50-482

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 221
License No. NPF-42

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

   A. The application for amendment to the Wolf Creek Generating Station, Unit 1 (the facility) Renewed Facility Operating License No. NPF-42 filed by the Wolf Creek Nuclear Operating Corporation (the Corporation), dated January 17, 2017, as supplemented by letters dated March 22, May 4, July 13, October 18, and November 14, 2017; January 15, January 29, April 19, June 19, August 9, November 15 (two letters), and December 6, 2018; and March 5, May 2, and May 5, 2019, complies 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 application, as amended, 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.

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 Renewed Facility Operating License No. NPF-42 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 221, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented during startup (prior to entry into Mode 2) from Refueling Outage 23.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License and
Technical Specifications

Date of Issuance: May 31, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 221 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK GENERATING STATION, UNIT 1

DOCKET NO. 50-482

Replace the following pages of the Renewed Facility Operating License No. NPF-42 and 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.

Renewed Facility Operating License

REMOVE	INSERT
4	4

Technical Specifications

REMOVE	INSERT
1.1-2	1.1-2
1.1-3	1.1-3
1.1-4	1.1-4
1.1-5	1.1-5
1.1-6	1.1-6
2.0-1	2.0-1
--	3.1-21
--	3.1-22
3.3-7	3.3-7
3.3-8	3.3-8
3.3-9	3.3-9
3.3-10	3.3-10
3.3-11	3.3-11
3.3-15	3.3-15
3.3-16	3.3-16
3.3-17	3.3-17
3.3-51	3.3-51
3.3-52	3.3-52
3.3-53	3.3-53
3.4-1	3.4-1
3.4-4	3.4-4
3.4-42	3.4-42
3.4-43	3.4-43
3.7-3	3.7-3
3.7-26	3.7-26
3.7-27	3.7-27
3.7-33	3.7-33
3.7-34	3.7-34

Technical Specifications (continued)	
3.7-35	3.7-35
3.9-5	3.9-5
5-0-17	5.0-17
5-0-22	5.0-22
5-0-25	5.0-25
5-0-26	5.0-26

(5) The Operating Corporation, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) The Operating Corporation, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission, now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Operating Corporation is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 221, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Kansas Gas & Electric Company and Kansas City Power & Light Company shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Environmental Qualification (Section 3.11, SSER #4, Section 3.11, SSER #5)*

Deleted per Amendment No. 141.

*The parenthetical notation following the title of many license conditions denotes the section of the supporting Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

(continued)

1.1 Definitions (continued)

DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using the effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF 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. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

(continued)

1.1 Definitions (continued)

LEAKAGE (continued)	3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE); b. Unidentified LEAKAGE All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; c. Pressure Boundary LEAKAGE LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.
MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE--OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, of the USAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.

(continued)

1.1 Definitions (continued)

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates and the power operated relief valve lift settings and the Low Temperature Overpressure Protection (LTOP) System arming temperature, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6.
QUADRANT POWER TILT RATIO (QPTR)	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.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3565 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.

(continued)

1.1 Definitions (continued)

SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing all slave relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include, a continuity check of associated required testable actuation devices. The SLAVE RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the necessary accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.13 for the ABB-NV DNB correlation, and ≥ 1.18 for the WLOP DNB correlation.

2.1.1.2 The peak centerline temperature shall be maintained ≤ 5080 °F, decreasing by 58 °F per 10,000 MWD/MTU of burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.9 RCS Boron Limitations < 500°F

LCO 3.1.9 The boron concentration of the Reactor Coolant System (RCS) shall be greater than the all rods out (ARO) critical boron concentration.

APPLICABILITY: MODE 2 with k_{eff} < 1.0 with any RCS cold leg temperature < 500°F and with Rod Control System capable of rod withdrawal,
MODE 3 with any RCS cold leg temperature < 500°F and with Rod Control System capable of rod withdrawal,
MODES 4 and 5 with Rod Control System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RCS boron concentration not within limit.	A.1	Initiate boration to restore RCS boron concentration to within limit.	Immediately
	OR		
	A.2	Initiate action to place the Rod Control System in a condition incapable of rod withdrawal.	Immediately
	OR		
	A.3	----------NOTE---------- Not applicable in MODES 4 and 5. ----------	
		Initiate action to increase all RCS cold leg temperatures to ≥ 500°F,	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.9.1	Verify RCS boron concentration is greater than the ARO critical boron concentration.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
P. One or more Turbine Stop Valve Closure Turbine Trip channel(s) inoperable.	P.1 Place channel(s) in trip. OR P.2 Reduce THERMAL POWER to < P-9.	72 hours 76 hours
Q. One train inoperable.	----------NOTE---------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ---------- Q.1 Restore train to OPERABLE status. OR Q.2 Be in MODE 3.	 24 hours 30 hours
R. One RTB train inoperable.	----------NOTE---------- One train may be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE. ---------- R.1 Restore train to OPERABLE status. OR R.2 Be in MODE 3.	 24 hour 30 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
S. One or more required channel(s) inoperable.	S.1	Verify interlock is in required state for existing unit conditions.	1 hour
	OR		
	S.2	Be in MODE 3.	7 hours
T. One or more required channel(s) or train inoperable.	T.1	Verify interlock is in required state for existing unit conditions.	1 hour
	OR		
	T.2	Be in MODE 2.	7 hours
U. One trip mechanism inoperable for one RTB.	U.1	Restore inoperable trip mechanism to OPERABLE status.	48 hours
	OR		
	U.2	Be in MODE 3.	54 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
V. One channel inoperable.	----------NOTE---------- The inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels. ----------	
	V.1 Place channel in trip.	72 hours
	OR	
	V.2.1 B in MODE 2 with k_{eff} < 1.0.	78 hours
	AND	
	V.2.2.1 Initiate action to fully insert all rods.	78 hours
	AND	
	V.2.2.2 Initiate action to place the Rod Control System in a condition incapable of rod withdrawal.	78 hours
	OR	
	V.2.3 Initiate action to borate the RCS to greater than all rods out (ARO) critical boron concentration.	78 hours
W. One channel inoperable.	----------NOTE---------- The inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels. ----------	
	W.1 Place channel in trip.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
X. Required Action and associated Completion Time of Condition W not met. OR Two or more channels inoperable.	X.1.1 Initiate action to fully insert all rods. AND X.1.2 Initiate action to place the Rod Control System in a condition incapable of rod withdrawal. OR X.2 Initiate action to borate the RCS to greater than all rods out (ARO) critical boron concentration.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

----------NOTE----------
Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	----------NOTES---------- Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. --- Compare results of calorimetric heat balance calculation to power range channel output. Adjust power range channel output if calorimetric heat balance calculation results exceed power range channel output by more than + 2% RTP.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.3	----------NOTES---------- Not required to be performed until 24 hours after THERMAL POWER is ≥ 50% RTP. ---------- Compare results of the core power distribution measurements to Nuclear Instrumentation System (NIS) AFD. Adjust NIS channel if absolute difference is ≥ 3%.	31 effective full power days (EFPD)
SR 3.3.1.4	----------NOTE---------- This Surveillance must be performed on the reactor trip bypass breaker for the local manual shunt trip only prior to placing the bypass breaker in service. ---------- Perform TADOT.	62 days on a STAGGERED TEST BASIS
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.1.6	----------NOTE---------- Not required to be performed until 72 hours after achieving equilibrium conditions with THERMAL POWER ≥ 75 % RTP. ---------- Calibrate excore channels to agree with core power distribution measurements.	92 EFPD
SR 3.3.1.7	----------NOTES---------- 1. Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. 2. Source range instrumentation shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. ---------- Perform COT.	184 days

(continued)

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE [a]
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA
	3[b], 4[b], 5[b]	2	C	SR 3.3.1.14	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 112.3% RTP
b. Low	1[c], 2[f]	4	V	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 28.3% RTP
	2[h], 3[i]	4	W, X	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 28.3% RTP
3. Power Range Neutron Flux Rate					
a. High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 6.3% RTP with time constant ≥ 2 sec
b. High Negative Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤6.3% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1[c], 2[d]	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 35.3% RTP

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(c) Below the P-10 (Power Range Neutron Flux) interlock.
(d) Above the P-6 (Intermediate Range Neutron Flux) interlock.
(f) With $k_eff \geq 1.0$.
(h) With $k_eff < 1.0$, and all RCS cold leg temperatures ≥ 500° F, and RCS boron concentration ≤ the rods out (ARO) critical boron concentration, and Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(i) With all RCS cold leg temperatures ≥ 500° F, and RCS boron concentration ≤ the ARO critical boron concentration, and Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE [a]
5. Source Range Neutron Flux	2[e]	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.6 E5 cps
	3[b], 4[b], 5[b]	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.6 E5 cps
6. Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 1 (Page 3.3-19)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 2 (Page 3.3-20)
8. Pressurizer Pressure					
a. Low	1[g]	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 1930 psig
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2395 psig
9. Pressurizer Water Level - High	1[g]	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.9% of instrument span
10. Reactor Coolant Flow - Low	1[g]	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 88.9% of normalized flow

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlock.
(g) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.1-1 (page 3 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE [a]
11. Not Used.					
12. Undervoltage RCPs	1[g]	2/bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 10355 Vac
13. Underfrequency RCPs	1[g]	2/bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 57.1 Hz
14. Steam Generator (SG) Water Level Low-Low [l]	1,2	4 per gen	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 22.3% of Narrow Range Instrument Span
15. Not Used.					
16. Turbine Trip					
a. Low Fluid Oil Pressure	1[j]	3	O	SR 3.3.1.10 SR 3.3.1.15	≥ 534.20 psig
b. Turbine Stop Valve Closure	1[j]	4	P	SR 3.3.1.10 SR 3.3.1.15	≥ 1% open
17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	Q	SR 3.3.1.14	NA
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2[e]	2	S	SR 3.3.1.11 SR 3.3.1.13	≥ 6E-11 amp
b. Low Power Reactor Trips Block, P-7	1	1 per train	T	SR 3.3.1.5	NA
c. Power Range Neutron Flux, P-8	1	4	T	SR 3.3.1.11 SR 3.3.1.13	≤ 51.3% RTP

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(g) Above the P-7 (Low Power Reactor Trips Block) interlock.
(l) The applicable MODES for these channels are more restrictive in Table 3.3.2-1. (See Function 6.d.)
(j) Above the P-9 (Power Range Neutron Flux) interlock.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time for Condition A, B or C not met in MODE 1, 2, 3, or 4.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours
E. Required Action and associated Completion Time for Condition A, B or C not met during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	E.1 Suspend CORE ALTERATIONS. AND E.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

----------NOTE----------

Refer to Table 3.3.7-1 to determine which SRs apply for each CREVS Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.2	Perform COT.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.7.3	----------NOTE---------- The continuity check may be excluded. ---------- Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.7.4	----------NOTE---------- Verification of setpoint is not required. ---------- Perform TADOT.	18 months
SR 3.3.7.5	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.7.6	----------NOTE---------- Radiation monitor detectors are excluded from response time testing. ---------- Verify Control Room Ventilation Isolation ESF RESPONSE TIMES are within limits.	18 months on a STAGGERED TEST BASIS

Table 3.3.7-1 (page 1 of 1)
CREVS Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Manual Initiation	1, 2, 3, 4, (a) and (c)	2	SR 3.3.7.4	NA
2.	Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	1, 2, 3, 4, (a) and (c)	2 trains	SR 3.3.7.3 SR 3.3.7.6	NA
3.	Control Room Radiation-Control Room Air Intakes	1, 2, 3, 4, (a) and (c)	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.5 SR 3.3.7.6	(b)
4.	Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a, for all initiation functions and requirements.			

(a) During movement of irradiated fuel assemblies.
(b) Trip Setpoint concentration value ($\mu Ci/cm^3$) is to be established such that the actual submersion dose rate would not exceed 2 mR/hr in the control room.
(c) During CORE ALTERATIONS.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure is greater than or equal to the limit specified in the COLR;

b. RCS average temperature is less than or equal to the limit specified in the COLR; and

c. RCS total flow rate ≥ 361,200 gpm and greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 1.

----------NOTE----------
Pressurizer pressure limit does not apply during :

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- Not applicable to RCS total flow rate. ---------- One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.1.2	Verify RCS average temperature is less than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 361,200 gpm and greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.4	----------NOTE---------- Not required to be performed until 7 days after ≥ 95% RTP. ---------- Verify by precision heat balance that RCS total flow rate is ≥ 361,200 gpm and greater than or equal to the limit specified in the COLR.	 18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	----------NOTE---------- LCO 3.0.4c. is applicable. ----------	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ 60 μCi/gm.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT XE-133 not within limit OR DOSE EQUIVALENT I-131 > 60 μCi/gm.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	----------NOTE---------- Only required to be performed in MODE 1. ---------- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 500 μCi/gm.	7 days
SR 3.4.16.2	----------NOTE---------- Only required to be performed in MODE 1. ---------- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus Maximum Allowable Power

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (% RTP)
4	70
3	51
2	31

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.10 Two CREVS trains shall be OPERABLE.

----------NOTE----------
The control room envelope (CRE) and control building envelope (CBE) boundaries may be opened intermittently under administrative controls that ensure the building boundary can be closed consistent with the safety analysis.

APPLICABILITY: MODES 1, 2, 3, and 4,
During CORE ALTERATIONS,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CREVS train inoperable for reasons other than Condition B.	A.1	Restore CREVS train to OPERABLE status.	7 days
B. One or more CREVS trains inoperable due to an inoperable CRE boundary or an inoperable CBE boundary in MODES 1, 2, 3, or 4.	B.1	Initiate action to implement mitigating actions.	Immediately
	AND		
	B.2	Verify mitigating actions to ensure CRE occupant radiological exposures will not exceed limits and CRE occupants are protected from chemical and smoke hazards.	24 hours
	AND		
	B.3	Restore CRE boundary and CBE boundary to OPERABLE status.	90 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	D.1	Place OPERABLE CREVS train in CRVIS mode.	Immediately
	OR		
	D.2.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	D.2.2	Suspend movement of irradiated fuel assemblies.	Immediately
E. Two CREVS trains inoperable during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	E.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
OR	E.2	Suspend movement of irradiated fuel assemblies.	Immediately
One or more CREVS trains inoperable due to an inoperable CRE boundary or an inoperable CBE boundary during movement of irradiated fuel assemblies or during CORE ALTERATIONS.			

(continued)

3.7 PLANT SYSTEMS

3.7.13 Emergency Exhaust System (EES)

LCO 3.7.13 Two EES trains shall be OPERABLE.

----------NOTE----------
The auxiliary building or fuel building boundary may be opened intermittently under administrative controls that ensure the building boundary can be closed consistent with the safety analysis.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies in the fuel building.

----------NOTE----------
The SIS mode of operation is required only in MODES 1, 2, 3, and 4. The FBVIS mode of operation is required only during movement of irradiated fuel assemblies in the fuel building.

ACTIONS

----------NOTE----------
LCO 3.0.3 is not applicable to the FBVIS mode of operation.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One EES train inoperable.	A.1 Restore EES train to OPERABLE status.	7 days
B. Two EES trains inoperable due to inoperable auxiliary building boundary in MODE 1, 2, 3, or 4.	B.1 Initiate actions to implement mitigating actions. AND	Immediately (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Verify mitigating actions ensure main control room occupants do not exceed 10 CFR 50 Appendix A GDC 19 limits.	24 hours
	AND		
	B.3	Restore building boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4. OR Two EES trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the fuel building.	D.1	Place OPERABLE EES train in operation in FBVIS mode.	Immediately
	OR		
	D.2	Suspend movement of irradiated fuel assemblies in the fuel building.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two EES trains inoperable for reasons other than Condition B during movement of irradiated fuel assemblies in the fuel building.	E.1 Suspend movement of irradiated fuel assemblies in the fuel building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each EES train for ≥ 15 continuous minutes with the heaters operating.	31 days
SR 3.7.13.2	Perform required EES filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.3	Verify each EES train actuates on an actual or simulated actuation signal.	18 months

(continued)

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts, or if open, capable of being closed;

b. One door in the emergency air lock closed and one door in the personnel air lock capable of being closed; and

----------NOTE----------
An emergency personnel escape air lock temporary closure device is an acceptable replacement for an emergency air lock door.
---

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge Isolation valve.

----------NOTE----------
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls that ensure the building boundary can be closed consistent with the safety analysis.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in ≥ 0.1 rem TEDE to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in the following outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

   a. Reactor Makeup Water Storage Tank
   b. Refueling Water Storage Tank
   c. Condensate Storage Tank, and
   d. Outside Temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

(continued)

5.5 Programs and Manuals

5.5.18 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE, CRE boundary, control building envelope (CBE), and CBE boundary.

b. Requirements for maintaining the CRE and CBE boundary in their design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE and CBE boundaries in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

The following are exceptions to Section C.1 and C.2 of Regulatory Guide 1.197, Revision 0:

1. The Tracer Gas Test based on the Brookhaven National Laboratory Atmospheric Tracer Depletion (ATD) Method is used to determine the unfiltered air inleakage past the CRE and CBE boundaries. The ATD Method is described in WCNOC letters dated February 21, 2005 (WO 05-0003), June 29, 2007 (WM 07-0057), and September 28, 2007 (ET 07-0045).

d. Measurement, at designated locations, of the CRE pressure relative to the outside atmosphere during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the periodic assessment of the CRE boundary.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Specification 3.1.3: Moderator Temperature Coefficient (MTC),
2. Specification 3.1.5: Shutdown Bank Insertion Limits,
3. Specification 3.1.6: Control Bank Insertion Limits,
4. Specification 3.2.3: Axial Flux Difference,
5. Specification 3.2.1: Heat Flux Hot Channel Factor, $F_Q(Z)$,
6. Specification 3.2.2: Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$),
7. Specification 3.9.1: Boron Concentration,
8. SHUTDOWN MARGIN for Specification 3.1.1 and 3.1.4, 3.1.5, 3.1.6, and 3.1.8,
9. Specification 3.3.1: Overtemperature ΔT and Overpower ΔT Trip Setpoints,
10. Specification 3.4.1: Reactor Coolant System pressure, temperature, and flow DNB limits, and
11. Specification 2.1.1: Reactor Core Safety Limits.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-11397-P-A, "Revised Thermal Design Procedure."
2. WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification."
3. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)."

5. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON."

6. WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology."

7. WCAP 10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code."

8. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report."

9. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™."

10. WCAP-8745-P-A, "Design Bases for the Thermal Power ΔT and Thermal Overtemperature ΔT Trip Functions."

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 221 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION, UNIT 1

DOCKET NO. 50-482

1.0 INTRODUCTION

By letter dated January 17, 2017 (Reference 1), as supplemented by letters dated March 22, May 4, July 13, October 18, and November 14, 2017; January 15, January 29, April 19, June 19, August 9, November 15 (two letters), and December 6, 2018; and March 5, May 2, and May 15, 2019 (References 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, and 86, respectively), Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) requested changes to the Technical Specifications (TSs) for Wolf Creek Generating Station, Unit 1 (Wolf Creek or WCGS).

The amendment would revise the Wolf Creek TSs to replace the existing methodology for performing core design, non-loss-of-coolant-accident (non-LOCA) and loss-of-coolant-accident (LOCA) safety analyses with standard analysis methodologies that were developed by Westinghouse Electric Corporation (Westinghouse) then approved by the U.S. Nuclear Regulatory Commission (NRC or the Commission).[1] In addition, the proposed amendment would revise the Wolf Creek licensing basis by adopting the alternative source term (AST) radiological analysis methodology described in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident source term." This amendment request is for a full-scope implementation of the AST as described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (Reference 17).

On July 5, 2017, the NRC staff published a proposed no significant hazards consideration (NSHC) determination in the *Federal Register* (82 FR 31084) for the proposed amendment. Subsequently by letters dated July 13, October 18, and November 14, 2017; and January 15, January 29, April 19, June 19, and August 9, 2018, the licensee provided additional information that expanded the scope of the amendment request as originally noticed in the *Federal Register*. Accordingly, the NRC published a second proposed NSHC determination in the

[1] Westinghouse WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology," March 1978 (Reference 22).

Federal Register on October 2, 2018 (83 FR 49590), which superseded the original notice in its entirety. Supplemental letters dated November 15 (two letters) and December 6, 2018; and March 5, May 2, and May 15, 2019, provided additional information that clarified the application, did not expand the scope of the application as noticed on October 2, 2018, and did not change the NRC staff's proposed NSHC determination published in the *Federal Register* dated October 2, 2018.

2.0 REGULATORY EVALUATION

The amendment would revise the Wolf Creek TSs to replace the existing methodology for performing core design, non-LOCA and LOCA safety analyses with standard analysis methodologies that were developed by Westinghouse. In addition, the proposed amendment would revise the Wolf Creek licensing basis by adopting the AST radiological analysis methodology described in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67.

Regulations at 10 CFR 50.90 state that whenever a holder of a license wishes to amend the license, including technical specifications in the license, an application for amendment must be filed, fully describing the changes desired. Under 10 CFR 50.92(a) determinations on whether to grant an applied-for license amendment are to be guided by the considerations that govern the issuance of initial licenses or construction permits to the extent applicable and appropriate. Both the common standards for licenses and construction permits in 10 CFR 50.40(a) and those specifically for issuance of operating licenses in 10 CFR 50.57(a)(3), provide that there must be 'reasonable assurance' that the activities at issue will not endanger the health and safety of the public.

The categories of items required to be in the TSs are provided in 10 CFR 50.36(c). As required by 10 CFR 50.36(c)(2), "Limiting conditions for operation," the TSs will include limiting conditions for operation (LCOs), which, per 10 CFR 50.36(c)(2)(i), "are the lowest functional capability or performance levels of equipment required for safe operation of the facility." Per 10 CFR 50.36(c)(2)(i), "[w]hen a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met. The remedial actions, the operating procedures, the facility and equipment, the use of the facility, and other TSs, must collectively provide reasonable assurance that the applicant will comply with the Commission's regulations, and that the health and safety of the public will not be endangered.

The regulation at 10 CFR 50.36(c)(3), "Surveillance requirements," requires TSs to include items in the category of SRs, which are requirements "relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." Also, 10 CFR 50.36(a)(1) states that "[a] summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications."

The NRC staff's guidance for review of TSs is in Chapter 16, Section 16.0, Revision 3, "Technical Specifications," of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (hereafter, referred to as the SRP), dated March 2010 (Reference 18.j). As described therein, as part of the regulatory standardization effort, the NRC staff has prepared Standard Technical Specifications (STSs) for each of the LWR nuclear designs. NUREG-1431, "Standard Technical

Specifications – Westinghouse Plants," Revision 4, Volume 1, Specifications, dated April 2012 (Reference 19), contains the STSs for Westinghouse-designed plants.

2.1 Transition to Westinghouse Core Design and Safety Analyses

The NRC staff's review of the SRs covers fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, anticipated operational occurrences (AOOs), and postulated accidents. The NRC's acceptance criteria are based on the guidance in SRP, Section 4.2, "Fuel System Design"; and Chapter 15, Section 15.0, "Transient and Accident Analysis (References 18.b and 18.g).

As described in Sections 1.2.3, "Principal Design Criteria," and 3.1, "Conformance with NRC General Design Criteria," of the Wolf Creek Updated Final Safety Analysis Report (UFSAR) (Reference 20), the plant was designed in accordance with the following 10 CFR Part 50, Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants":

- Criterion 10, "Reactor Design," which requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs.

- Criterion 15, "Reactor coolant system design," requires that "[t]he reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

- Criterion 20, "Protection system functions," requires that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs, and to initiate automatically, operation of systems and components important to safety under accident conditions.

- Criterion 25, "Protection system requirements for reactivity control malfunctions," requires that "[t]he protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems...."

- Criterion 27, "Combined reactivity control system capability," requires that "[t]he reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained."

- Criterion 28, "Reactivity limits," requires the reactivity control system to be designed such that reactivity accidents cannot significantly damage the reactor coolant pressure boundary nor result in disturbance to the core, associated structure, or other internals that impedes cooling.

- Criterion 35, "Emergency core cooling," provides applicable emergency core cooling system (ECCS) design criteria, stating that the ECCS shall "transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad

> damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts."

The licensee is not proposing to change its design criteria. Accordingly, the above GDC were used by the NRC staff to evaluate as the staff determined how Wolf Creek continued to meet its unchanged design criteria.

Per 10 CFR 50.46(a)(1)(i) "Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in [10 CFR 50.46(b)]." Section 50.46(b)(5) of 10 CFR states:

> After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The methodology in Westinghouse WCAP-8339, "Westinghouse Emergency Core Cooling System Evaluation Model – Summary," dated June 1974 (Reference 21), for large-break LOCA (LBLOCA) analysis, ensures compliance with this criterion, in part, by demonstrating that the core will remain subcritical following a LOCA using only borated water. Therefore, the licensee performed an analysis that generated the post-LOCA containment sump boron concentration as a function of the pre-LOCA reactor coolant system (RCS) boron concentration. This analysis is then used in the development of a core reactivity limit that is confirmed to be met each cycle as part of the Westinghouse Reload Safety Evaluation Methodology as described in WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology," March 1978 (Reference 22).

The licensee also sought to demonstrate compliance with 10 CFR 50.46(b)(5) by performing a long-term core cooling analysis showing that the ECCS provides sufficient core decay heat removal capability following the initial phases of the LOCA. The licensee provided analyses demonstrating that there is sufficient ECCS injection to remove decay heat when the ECCS suction is transferred from the refueling water storage tank (RWST) to the containment sump and when a portion of the ECCS injection is transferred from the cold legs to the hot and cold legs. The licensee's analysis also established a time for hot leg recirculation to avoid boric acid precipitation (BAP), which could potentially impede the capability for decay heat removal.

Specific review criteria for fuel are contained in SRP, Section 4.2 (Reference 18.b), which provides guidance for the NRC staff's review regarding fuel system design. As stated in SRP, Section 4.2, the objectives of the fuel system review are to provide assurance that:

(1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs),

(2) fuel system damage is never so severe as to prevent control rod insertion when it is required,

(3) the number of fuel rod failures is not underestimated for postulated accidents, and

(4) coolability is always maintained.

For the non-LOCA accidents, the results of the analyses are reviewed to ensure that pertinent system parameters are within expected ranges. SRP, Section 15.0 (Reference 18.g), for a non-LOCA, identifies parameters of importance that include:

- RCS pressure
- Steam generator (SG) pressure, fluid temperatures
- Clad temperatures
- Steam line and feedwater flow rates
- Pressurizer and SG water levels
- Reactor power
- Total core reactivity
- Hot and average channel heat flux
- Minimum departure from nucleate boiling ratio.

The licensee reanalyzed the non-LOCA events for Wolf Creek and established that the applicable acceptance criteria are met for each event. The details of this conclusion are discussed in WCAP-17658-NP, Revision 1-C, "Wolf Creek Generating Station Transition of Methods for Core Design and Safety Analyses – Licensing Report" (see Enclosure I of the letter dated January 17, 2017 (Reference 1))[2].

2.2 Adoption of AST

In the past, power reactor licensees have typically used U.S. Atomic Energy Commission Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962[3] (Reference 25), as the basis for DBA analysis source terms. The regulation in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," which contains offsite dose limits in terms of whole body and thyroid dose, makes reference to TID-14844.

Section 50.67 of 10 CFR, provides a mechanism for licensed power reactors to replace the traditional accident source term (TID-14844) used in their design-basis accident (DBA) analyses with an AST. Regulatory guidance for the implementation of these ASTs is provided in RG 1.183. Section 50.67 of 10 CFR requires that a licensee seeking to use an AST, apply for a license amendment and include an evaluation of the consequences of the affected DBAs. The licensee's application dated January 17, 2017 (Reference 1), as supplemented, addresses these requirements in proposing to use the AST, described in RG 1.183, as the Wolf Creek DBA source term used to evaluate the radiological consequences of DBAs. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and GDC 19 of Appendix A to 10 CFR Part 50 for several DBAs as discussed below.

The accident source term is intended to be representative of a major accident involving significant core damage and is typically postulated to occur in conjunction with a LBLOCA. As a result of significant core damage, fission products are available for release into the containment environment. An AST is an accident source term that is different from the accident source term

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[2] Referred to as Enclosure I, thereafter.
[3] Referred to as TID-14844, thereafter.

used in the original design and licensing of the facility and has been approved for use under 10 CFR 50.67. Although an acceptable AST is not set forth in the regulations, RG 1.183 identifies an AST that is acceptable to the NRC staff for use at operating reactors.

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements on which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67; the accident specific guideline values in Regulatory Position 4.4, "Acceptance Criteria," of RG 1.183; and Table 1 of NUREG-0800, Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000 (Reference 18.h). Except where the licensee has proposed an acceptable alternative, the NRC staff used the regulatory guidance in the documents listed below in doing this review.

In 10 CFR 50.36, "Technical specifications," the Commission established its regulatory requirements related to the content of TSs. TSs are required to include items in, among others, the following five categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls. The proposed changes to the TSs represent changes to the existing TSs LCOs, SRs, and administrative controls and addition of a new LCO and associated SRs.

The licensee's request is pursuant to 10 CFR 50.67, which provides a mechanism for licensed power reactors to replace the traditional source term used in the radiological consequence analyses of DBAs. The licensee's current DBA radiological consequence analyses are based on the source term from TID-14844.

The NRC staff evaluated the licensee's analysis of the radiological consequences of the affected DBAs for implementation of the AST methodology and the associated changes to the TSs proposed by the licensee against the radiological dose requirements specified in 10 CFR 50.67(b)(2) and the dose limits specified in GDC 19 of Appendix A to 10 CFR Part 50." Paragraph 50.67(b)(2) of 10 CFR requires that the licensee's analysis demonstrates with reasonable assurance that:

i. An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv [Sieverts] (25 rem [roentgen equivalent man]) total effective dose equivalent (TEDE).

ii. An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

iii. Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

This SE addresses the impact of the proposed changes on radiological consequences of the previously analyzed DBAs and the acceptability of the revised analysis results. The regulatory

requirements from which the NRC staff based its acceptance are the accident radiation dose values in 10 CFR 50.67; the accident specific guideline values in Regulatory Position 4.4 of RG 1.183; and Table 1 of SRP, Section 15.0.1, Revision 0, "Radiological Consequence Analyses Using Alternative Source Terms," dated July 2000 (Reference 18.h), unless otherwise discussed below.

In addition, Table 3 of RG 1.183 provides gap release fractions that have been found acceptable for use by the NRC staff for non-LOCA events. Footnote 11 in Table 3 provides the conditions under which these gap fractions have been found to be acceptable, and states:

> The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU [megawatt days per metric ton of uranium] provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft [kilowatt per foot] peak rod average power for burnups exceeding 54 GWD/MTU [gigawatt days per metric ton of uranium]. As an alternative, fission gas release calculations performed using NRC approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR [boiling-water reactor rod drop accident and the PWR [pressurized water reactor] rod ejection accident, the gap fractions are assumed to be 10 percent for iodines and noble gases.

Revised gap release fractions were calculated by the Pacific Northwest National Laboratory (PNNL) and NRC staff in PNNL Report 18212, "Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS [American NuclearSociety] 5.4 Standard," Revision 1, June 2011 (Reference 24).

The NRC staff's evaluation for adoption of the AST, including atmospheric dispersion factors (χ/Q) (also known as relative concentrations) used in radiological dose analyses for implementation of full-scope AST and human factors considerations for credited manual operator actions, is based upon the following regulations, regulatory guides, and standards:

- The regulations at 10 CFR 50.34, "Contents of applications; technical information," define the content requirements for the UFSAR, including evaluations required to show that accident dose criteria are met.

- The regulations at 10 CFR 50.67, "Accident source term," provide a mechanism for licensed power reactors to replace the traditional source term used in the radiological consequence analyses of DBAs.

- Criterion 19, "Control room," of Appendix A to 10 CFR Part 50 requires that, "holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident."

- RG 1.23, Revision 1, "Meteorological Monitoring Programs for Nuclear Power Plants," dated March 2007 (Reference 26).

- RG 1.24, Revision 0, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure," dated March 1972 (Reference 27).

- RG 1.52, Revision 3, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water Cooled Nuclear Power Plants," June 2001 (Reference 28).

- RG 1.145, Revision 1, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," dated November 1992 (Reference 29).

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (Reference 17).

- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," dated June 2003 (Reference 30).

- RG 1.196, Revision 1, "Control Room Habitability at Light-Water Nuclear Power Reactors," dated January 2007 (Reference 31).

- RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," dated May 2003 (Reference 32).

- The following sections of NUREG-0800:

    - Section 2.3.4, "Short-Term Atmospheric Dispersion Estimates for Accident Releases," Revision 3 (Reference 18.a);

    - Section 6.4, "Control Room Habitability System," Revision 3 (Reference 18.c);

    - Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 4 (Reference 18.d);

    - Section 11.3, "Gaseous Waste Management System," Revision 4 (Reference 18.f);

    - Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0 (Reference 18.h);

    - Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside the Containment," Revision 2 (Reference 18.i);

    - Section 18.0, "Human Factors Engineering," and Attachment A, "Guidance for Evaluating Credited Manual Operator Actions," Revision 3 (Reference 18.k); and

    - Branch Technical Position 11-5, Revision 4, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," Revision 4 (Reference 18.l).

- NUREG-0696, "Functional Criteria for Emergency Response Facilities," Final Report, dated February 1981 (Reference 33).

- NUREG-0737, "Clarification of TMI [Three Mile Island Nuclear Station] Action Plan Requirements," dated November 1980 and "Clarification of TMI Action Plan Requirements - Requirements for Emergency Response Capability," Supplement No. 1, Reprinted February 1989 (Reference 34).

- Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms," dated March 2006 (Reference 35).

The Guidance in RG 1.183 was also used to review the licensee's analysis to ensure that suppression pool pH (scale of acidity) is maintained greater than or equal to (≥) 7 for 30 days following a LOCA to minimize re-evolution of iodine from the suppression pool water.

2.3 Requirements for Instrumentation and Control Systems for Transition to Westinghouse Core Design and Safety Analyses and Adoption of AST

The applicable portions of the following regulatory requirements and guidance was considered by the NRC staff for the review of TS changes related to instrumentation and control systems for transition to Westinghouse core design and safety analyses and adoption of AST:

- Section 50.55a(h), "Protection and safety systems," of 10 CFR requires that the protection systems must meet the requirements in Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 279–1968, "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems," or the requirements in IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," or the requirements in IEEE Std. 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995.

- Section 4.1, "Applicable Regulatory Requirements/Criteria," of Attachment I to the license amendment request (LAR) dated January 17, 2017, describes the applicable GDC of Appendix A to 10 CFR Part 50, and provides an assessment of conformance for each of these criteria included as the plants licensing basis. The licensee is not proposing to change its design criteria (with the exception of the aspect of Criterion 19 that specifically addresses "holders of operating licenses using an alternative source term under § 50.67"). Accordingly, the following GDC were used by the NRC staff to evaluate as the staff determined how Wolf Creek continued to meet the unchanged design criteria.

    - Criterion 10, "Reactor design," requires that "[t]he reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

    - Criterion 13, "Instrumentation and control," requires that "[i]nstrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables

and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges."

- Criterion 15, "Reactor coolant system design," requires that "[t]he reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

- Criterion 19, "Control room," of Appendix A to 10 CFR Part 50 requires that, "holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident."

- Criterion 20, "Protection system functions," requires that "[t]he protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety."

- Criterion 21, "Protection system reliability and testability," requires, in part that "[t]he protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed."

- Criterion 22, "Protection system independence," requires, in part that "[t]he protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis."

- Criterion 25, "Protection system requirements for reactivity control malfunctions," requires that "[t]he protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods."

- Criterion 26, "Reactivity control system redundancy and capability," requires that "[t]wo independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power

changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions."

- SRP, Chapter 7, Section 7.0, Revision 7, "Instrumentation and Controls – Overview of Review Process" (Reference 18.e), defines the acceptance criteria for this review. Section 7.0 addresses the requirements for instrumentation and control systems in light-water nuclear power plants.

- RG 1.105, "Setpoints for Safety Related Instrumentation," Revision 3 (Reference 36), describes a method acceptable to the NRC staff for complying with the NRC's regulations for ensuring that instrumentation setpoints are initially within, and remain within the TS limits. The RG endorses Part I of International Society of Automation (ISA) Standard ISA-S67.04-1994, "Setpoints for Nuclear Safety Instrumentation" (Reference 37) subject to the NRC staff clarifications.

2.4 Proposed Technical Specification Changes

2.4.1 Transition to Westinghouse Core Design and Safety Analyses

a. TS Section 2.1.1, "Reactor Core SLs [Safety Limits]"

TS 2.1.1.1 will be revised to change the values for departure from nucleate boiling ratio. The existing TS 2.1.1.1 states:

> The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.23 for the WRB-2 DNB [departure from nucleate boiling] correlation, and ≥ 1.30 for the W-3 DNB correlation.

Revised TS 2.1.1.1 would state:

> The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.13 for the ABB-NV DNB correlation, and ≥ 1.18 for the WLOP [Westinghouse low pressure] DNB correlation.

b. New TS 3.1.9, "RCS Boron Limitations < 500 °F"

New LCO 3.1.9 will be added to specify that the boron concentration of the RCS shall be greater than the all rod out (ARO) critical boron concentration, including Applicability, Actions (Conditions, Required Actions, and Completion Times,) and SRs.

c. TS 3.3.1, "Reactor Trip System (RTS) Instrumentation"

New Conditions V, W, and X, including Required Actions and Completion Times will be added to TS 3.3.1 to address the impact of change to TS Table 3.3.1-1, "Reactor Trip System Instrumentation," Function 2, "Power Range Neutron Flux."

d. TS Table 3.3.1-1, "Reactor Trip System Instrumentation"

In TS Table 3.3.1-1, the applicable Conditions and SRs will be revised for Function 2.b, "Power Range Neutron Flux," and new Footnotes f, h, and i are added to reflect changes to TS 3.3.1 described in item c above. Also, the allowable value for Function 10, "Reactor Coolant Flow - Low," will be revised from ≥ 88.9% of design flow (90,324 gallons per minute (gpm)) to ≥ 88.9% of normalized flow.

e. TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"

For the RCS total flow rate limit, one of the RCS DNB parameters, specified in LCO 3.4.1.c, will be revised from ≥ 37.1×10^4 gpm to ≥ 361,200 gpm and greater than or equal to the limit specified in the COLR [core operating limit report]. Also, SRs 3.4.1.3 and 3.4.1.4 will be revised to reflect the change in the RCS total flow rate.

f. Table 3.7.1-1, "OPERABLE Main Steam Safety Valves versus Maximum Allowable Power"

Maximum Allowable Power (as percent of the rated thermal power (RTP)) for a number of operable main steam safety valves (MSSVs) per SG will be revised as follows:

Number of Operable MSSVs per SG	Maximum Allowable Power (% RTP)
4	From 87 to 70
3	From 65 to 51
2	From 44 to 31

g. TS 5.6.5, "CORE OPERATING LIMIT REPORT (COLR)

TS 5.6.5 will be revised to delete existing Reference 1, WCNOC Topical Report TR 90-0025; Reference 3, WCNOC Topical Report NSAG-006; Reference 5, WCNOC Topical Report NSAG-007; and Reference 6, NRC Staff Safety Evaluation Report dated March 30, 1993, for the revision to TS for Cycle 7.

WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," will be added as new Reference 3. There is no other change to the references, except for renumbering due to deleted references.

2.4.2 TS Changes due to the Adoption of AST

a. TS 1.1, "Definitions"

Definition for Dose Equivalent I-131 [Iodine 131] will be revised to delete references to Table III of TID-14844 (Reference 25), Table E-7 of RG 1.109, Revision 1 (Reference 38), and International Commission on Radiological Protection (ICRP) Publication 30, "Limits for Intakes of Radionuclides by Workers," Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity" (Reference 39). Also, the definition of Dose Equivalent XE-133 [Xenon 133] will be revised to delete reference to dose conversion factors from Table B-1 of RG 1.109.

b. TS 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation"

LCO 3.3.7 states that "The CREVS actuation instrumentation for each Function in Table 3.3.7-1 shall be OPERABLE." The Applicability of this LCO is also specified in Table 3.3.7-1.

Condition E applies when the Required Action and Completion Time for Condition A, B, or C not met during movement of irradiated fuel assemblies. The Required Action is to suspend core alterations and movement of irradiated fuel assemblies.

Condition E currently states,

> Required Action and associated Completion Time for Condition A, B or C not met during movement of irradiated fuel assemblies.

Revised Condition E will state,

> Required Action and associated Completion Time for Condition A, B or C not met during movement of irradiated fuel assemblies **or during Core Alterations**.

Also, a new SR 3.3.7.6 will be added for verification of the control room ventilation isolation engineered safety feature (ESF) response times every 18 months on a staggered test basis.

c. TS Table 3.3.7-1, "CREVS Actuation Instrumentation"

Table 3.3.7-1 will be revised to reflect the change to Condition E of TS 3.3.7 in Item b above, by adding Footnote c and incorporating it in to the Applicable Modes or Other Specified Conditions for Function 1, "Manual Initiation"; Function 2, "Automatic Actuation Logic and Actuation Relays (BOP ESFAS[4])"; and Function 3, "Control Room Radiation – Control Room Air Intakes." Also, new SR 3.3.7.6 will be added to the SRs for Conditions 2 and 3 in TS Table 3.3.7-1.

d. TS 3.4.16, "RCS Specific Activity"

LCO 3.4.16 requires that the RCS dose equivalent I-131 and dose equivalent XE-133 specific activity shall be within limit. Condition B applies when dose equivalent XE-133 is not within limit. The Required Action is to restore dose equivalent XE-133 to within the limit within 48 hours.

The proposed change deletes existing Conditions B and associated Required Actions and Completion Time. Actions associated with Condition for Dose Equivalent XE-133 not within limit will be merged with existing Condition C. Existing Condition C will be numbered as Condition B.

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[4] Balance-of-Plant Engineered Safety Feature Actuation System (BOP ESFAS)

e. TS 3.7.10, "Control Room Emergency Ventilation System (CREVS)"

LCO 3.7.10 requires "[t]wo CREVS trains to be OPERABLE." The LCO is modified by a note which states:

> The control room envelope (CRE) and control building envelope (CBE) boundaries may be opened intermittently under administrative controls.

The LCO note is being revised and would state:

> The control room envelope (CRE) and control building envelope (CBE) boundaries may be opened intermittently under administrative controls that ensure the building boundary can be closed consistent with the safety analysis.

The LCO is currently applicable during MODES 1 (Power Operation), 2 (Startup), 3 (Hot Standby), and 4 (Hot Shutdown) and during movement of irradiated fuel assemblies. The LAR proposes to expand the applicability to include "during core alterations." Additionally, the applicability of Conditions D and E is modified to add "or during core alterations."

f. TS 3.7.13, "Emergency Exhaust System (EES)"

LCO 3.7.13 requires "[t]wo EES trains to be OPERABLE. The LCO is modified by a note which states:

> The auxiliary building or fuel building boundary may be opened intermittently under administrative controls.

LCO 3.7.13 note is being revised and would state:

> The auxiliary building or fuel building boundary may be opened intermittently under administrative controls that ensure the building boundary can be closed consistent with the safety analysis.

In addition, the Required Actions and Completion Times associated with Conditions B and E have been revised. Current TS 3.7.13 Condition B states:

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two EES trains inoperable due to inoperable auxiliary building boundary in MODE 1,2,3 or 4.	B.1 Restore auxiliary building boundary to OPERABLE status.	24 hours

Revised TS 3.7.13 Condition B would state:

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two EES trains inoperable due to inoperable auxiliary building boundary in MODE 1,2,3 or 4.	B.1 Initiate actions to implement mitigating actions. AND B.2 Verify mitigating actions ensure main control room occupants do not exceed 10 CFR 50 Appendix A GDC 19 limits. AND B.3 Restore building boundary to OPERABLE status	Immediately 24 hours 24 hours

Current TS 3.7.13 Condition E states:

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two EES trains inoperable due to inoperable fuel building boundary during movement of irradiated fuel assemblies in the fuel building.	B.1 Restore fuel building boundary to OPERABLE status.	24 hours

Revised TS 3.7.13 Condition E would state:

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two EES trains inoperable for reasons other than Condition B during movement of irradiated fuel assemblies in the fuel building.	B.1 Suspend movement of irradiated fuel assemblies in the fuel building.	Immediately

The current TS 3.7.13 Condition F applies when the Required Action and associated Completion Time of Condition E is not met or when two EES trains are inoperable during movement of irradiated fuel assemblies in the fuel building for reasons other than Condition E.

Condition F has been deleted as a result of changes to Condition E.

g. TS 3.9.4, "Containment Penetrations"

LCO 3.9.4 specifies the required status for containment penetrations during core alterations and during movement of irradiated fuel assemblies within containment.

LCO 3.9.4.c is modified by a Note which states:

> Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

LCO 3.9.4.c Note is being revised and would state:

Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls that ensure the building boundary can be closed consistent with the safety analysis.

h. TS 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program"

TS 5.5.12.b currently states:

> A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

TS 5.5.12.b will be revised and would state:

> A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in ≥ 0.1 rem TEDE to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

i. TS 5.5.18, "Control Room Envelope Habitability Program"

TS 5.5.18, introductory paragraph stating the requirements for establishing control room envelope habitability program currently states in part:

> ...The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

TS 5.5.18, introductory paragraph will be revised and would state, in part:

> ...The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident

(DBA) conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. The program shall include the following elements:

3.0 TECHNICAL EVALUATION

3.1 Instrumentation and Control Systems - Transition to Westinghouse Core Design and Safety Analyses and Adoption of AST

In the LAR, the licensee requested to modify its existing WCNOC methodology for performing core design, non-LOCA and LOCA safety analyses to the standard Westinghouse NRC-approved methodologies. This proposed change would result in revision of TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," Table 3.3.1-1, Function 10, "Reactor Coolant Flow – Low." In addition, this proposed amendment would add a new LCO 3.1.9, "RCS Boron Limitations < 500 °F [degree Fahrenheit]." This last modification would require modification to the requirements for TS Table 3.3.1-1, Function 2.b, "Power Range Neutron Flux - Low." Specifically, the Applicability for the RTS Trip Function 2.b will be revised and new Conditions V, W, and X will be added to LCO 3.3.1 for that RTS trip function.

In addition, the licensee requested to revise its licensing basis by adopting the AST radiological analysis methodology. As a result, the licensee is adding new SR 3.3.7.6 to TS 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation."

Below are detailed descriptions and evaluations of the proposed TS modifications.

3.1.1 TS Table 3.3.1-1, Function 10

For the transition to Westinghouse methodology for performing core design Non-LOCA and LOCA safety analyses, the licensee is proposing to modify the allowable value (AV) for TS Table 3.3.1-1, Function 10, "Reactor Coolant Flow – Low." The proposed change deletes Footnote "m" which identifies the "% of design flow - 90,324 gpm" portion of the AV. The current AV for Function 10 is defined as:

> "≥ 88.9% of design flow - 90,324 gpm"

The proposed change would revise the TS AV to:

> "≥ 88.9% of Normalized Flow"

The licensee is revising the AV for consistency with the assumptions made in the new safety analysis methodology, which used normalized flow instead of design flow.

Enclosure III, "WCAP-18083, Revision 0, "Westinghouse Revised Thermal Design Procedure Uncertainty Calculations for the Wolf Creek Generating Station," of the LAR dated January 17, 2017 (Reference 1),"[5] defines the "Normalized Flow as the RCS flow normalization to the RCS flow calorimetric. Wolf Creek TSs require a RCS flow calorimetric measurement to verify RCS flow at the beginning of every fuel cycle at or near 100 percent RTP operation. The flow calorimetric measurement is performed by determining the SG thermal output (corrected for the reactor coolant pump (RCP) heat input and the loop's share of primary system heat losses) and

[5] Referred to as Enclosure III, thereafter.

the enthalpy rise of the primary coolant. Enclosure III, Section 3.3, "RCS Flow Calorimetric Measurement Uncertainties," describes this calculation. The RCS flow is monitored by performing a RCS flow calorimetric measurement at the beginning of each cycle for verification of minimum measured flow (MMF) and every 12 hours for RCS flow degradation when in MODE 1 (required by SR 3.4.1.3). The loop RCS flow channels are normalized to the RCS flow calorimetric and are used to satisfy the 12-hour RCS flow requirements of SR 3.4.1.3. As part of the amendment, the licensee modifies the RCS total flow rate from 3.71×10^4 gpm to 3.76×10^4 gpm. The proposed RCS total flow will be used for adopting AST for Wolf Creek radiological analysis methodology. The NRC staff found an inconsistency with the RCS total flow rate identified in the submittal dated July 13, 2017 (Reference 4), and the value identified in the docketed and proposed TS. The licensee provided an explanation and revised the response in a letter dated January 29, 2018 (Reference 8). The licensee clarified that the current RCS total flow rate is 37.1×10^4 gpm, and it will be modified to 37.6×10^4 gpm for the limit specified in the COLR. Note that the proposed TS in the amendment proposes a modification of the RCS total flow rate of 361,200 gpm in TS 3.4.1, which represents the RCS thermal design flow. Acceptance of this modification is discussed in Section 3.6.6.3 of this SE.

In the responses to a request for additional information (RAI), provided by letter dated July 13, 2017 (Reference 4), the licensee clarified that there are two separate normalization processes. First the normalization process to establish the "Normalized Flow" to be listed in TS Table 3.3.1-1, Function 10. Second, the normalization process for the 12-hour surveillance test. The licensee also explained how the normalization process for the Reactor Coolant Flow - Low trip function and RCS flow are performed.

Enclosure III describes the calculations used to establish the normalized flow to be listed in Table 3.3.1-1, Function 10. These calculations were based on WCAP-17504-NP-A, Revision 1, "Westinghouse Generic Setpoint Methodology" (Reference 40). This enclosure also provides a brief description of the Westinghouse methodology as applied to Wolf Creek. Specifically, the Westinghouse setpoint methodology combines the uncertainty that is statistically and functionally independent and adds uncertainties considered dependent, which are then combined with the independent terms using the square root of the sum of the squares. This methodology is designed to result in a 95 percent probability and confidence level of providing a channel trip before the process variable reaches the analytical limit. However, in the amendment, the licensee noted that Westinghouse was not able to confirm that non-Westinghouse equipment that instruments uncertainties have been estimated at the 95/95 two-sided statistical level. The NRC staff did not review the data analyses for non-Westinghouse equipment because for the uncertainty calculations, Westinghouse used information from the plant baseline design input documentation to establish all appropriate and applicable uncertainties. Also, in these calculations, Westinghouse used conservative values and added additional margin to the uncertainty calculations.

Using its methodology, Westinghouse calculated the uncertainties, known as Channel Statistical Allowance, for the following parameters: pressurizer pressure, primary coolant temperature (T_{avg}), RCS calorimetric measurement, loop RCS flow, and the daily power calorimetric. These parameters are used in the uncertainty analysis of the revised thermal design procedure (RTDP). Enclosure III provides detail information, including assumptions and equations used to estimate the RCS flow calorimetric uncertainties. This enclosure also identifies the sources of information used in the uncertainty calculations and individual parameter uncertainties and instrument channel uncertainty Channel Statistical Allowance calculations for the RTDP functions.

The NRC staff reviewed the methodology and calculations provided in Enclosure III and found the uncertainties' terms and setpoint calculations acceptable for the proposed AV to meet the regulation in 10 CFR 50.36(c)(1)(ii)(A) regarding limiting safety system settings and limiting control settings. The revised WCGS TSs include requirements for automatic protective action to correct an abnormal situation or shutdown the reactor before a safety limit is exceeded. The staff did not review the data analyses for non-Westinghouse equipment because of the acceptable margin and conservative valves applied to the analyses.

3.1.2 TS Table 3.3.1-1, Function 2.b

The proposed amendment adds a new LCO 3.1.9, "RCS Boron Limitations < 500 °F." This new LCO requires that the RCS is borated to greater than the ARO critical boron concentration to provide sufficient shutdown margin if the rods are capable of being withdrawn in applicability MODES specified by the LCO.

Due to this addition, the requirements for RTS trip Function 2.b, Power Range Neutron Flux - Low, in TS Table 3.3.1-1 is modified. In this manner, RTS trip Function 2.b is available to mitigate an uncontrolled rod cluster control assembly (RCCA) bank withdrawal event postulated to occur during low power or subcritical (startup) conditions. Thus, the Applicability for RTS trip Function 2.b is modified and new Conditions V, W, and X is added to LCO 3.3.1 for Function 2.b. The licensee provided modified TS pages showing the wording proposed for these new conditions.

Also, the licensee added new Footnotes (f), (h), and (i) to the Applicability of RTS trip Function 2.b in TS Table 3.3.1-1 to reflect the revised Applicability requirements. These new footnotes are worded as follows:

(f) With k_{eff} [reactivity coefficient] ≥ 1.0.

(h) With k_{eff} < 1.0, and all RCS cold leg temperatures ≥ 500 °F, and RCS boron concentration ≤ the all rods out (ARO) critical boron concentration, and Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(i) With all RCS cold leg temperatures ≥ 500 °F, and RCS boron concentration ≤ the ARO critical boron concentration, and Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

In the current TS, Function 2.b includes all of MODE 2 and invokes Condition E for an inoperable channel. The new Footnote (f) divides the current MODE 2 Applicability for RTS trip Function 2.b into critical and subcritical portions. With the proposed modification, when critical in MODE 2, with $k_{eff} \geq 1.0$, per new Footnote (f), failure to meet the Required Channels of TS Table 3.3.1-1 for RTS trip Function 2.b will result in new Condition V entry. The new Condition V is similar to the current Condition E, and in that the required action for both is to either place the one inoperable channel in trip or require that the plant be placed in a condition where the trip function is not needed to protect the core from events applicable to the inoperable channel.

When the reactor is subcritical in MODE 2, with k_{eff} < 1.0 and the plant is meeting the specified conditions in new Footnote (h), failure to meet the required channels of TS Table 3.3.1-1 for RTS trip Function 2.b will require that new Condition W and Condition X, if applicable, be

entered. The Applicability for RTS trip Function 2.b is extended to the upper portion of MODE 3 with the plant meeting the specified conditions in new Footnote (i). New Conditions W and X also apply to this Applicability.

The licensee added these modifications to cover the hypothetical portion of MODE 2 with the reactor subcritical ($k_{eff} < 1.0$) and the upper portion of MODE 3. When the reactor is subcritical in MODE 2 or the plant is in the upper portion of MODE 3, Conditions W and X will cover situations where the required channels for RTS trip Function 2.b in Table 3.3.1-1 are not met. Thus, the Power Range Neutron Flux - Low trip Function ensures that protection is provided against an uncontrolled RCCA bank withdrawal during low power or subcritical (startup) conditions.

The NRC staff reviewed the proposed modifications for the RTS trip Function 2.b, Power Range Neutron Flux - Low, in TS Table 3.3.1-1, and found they were added to provide protection during an uncontrolled RCCA bank withdrawal from a low power or subcritical condition event. The staff found this to be acceptable because the new LCO provides adequate shutdown margin (by borating the RCS), or an automatic reactor trip is included to protect the plant (the Power Range Neutron Flux - Low trip Function 2.b would be available to provide the necessary protection should an RCCA bank withdrawal from a low power or subcritical (RWFS)[6] event occur). Based on this information, the staff determined that the new LCO provides adequate shutdown margin (by borating the RCS) or an automatic reactor trip to protect the plant. Therefore, the staff concludes that the WCGS instrumentation will continue to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. The criteria of GDC 13 and GDC 20 are therefore satisfied.

3.1.3 SR 3.3.7.6, Response Time Testing for Control Room Ventilation Isolation

As part of this amendment, the licensee proposed revising the Wolf Creek licensing basis by adopting AST for Wolf Creek radiological analysis methodology. Enclosure IV, "Full Scope Implementation of Alternative Source Term," of the LAR dated January 17, 2017 (Reference 1),[7] provides detailed description and evaluation associated with the implementation of the AST.

To support adoption of AST, the licensee proposed adding a new SR 3.3.7.6 to TS 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation." The new SR will test the components credited for the switch to emergency mode ventilation. Specifically, this SR will be applied to Functions 2 and 3 in Table 3.3.7-1.

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[6] Defined as Uncontrolled RCCA Bank Withdrawal from a Low Power or Subcritical Condition Event in Technical Specifications Task Force (TSTF)-453, Revision 2, "Addition of New [TS] on RCS Boron Limits and Revisions to [TS] 3.3.1 to Address RWFS.

[7] Referred to as Enclosure IV, thereafter.

The proposed SR would state:

> ----------NOTE----------
> Radiation monitor detectors are excluded from response time testing
> ----------
>
> Verify Control Room Ventilation Isolation ESF RESPONSE TIMES are within limits.

With a Frequency of "18 months on a STAGGERED TEST BASIS."

The licensee stated that this SR is necessary to verify the required response time of the channel. Further, this SR will ensure that the time used in the AST model is bounded. The revised TS pages for TS 3.3.7 do not include the limit for the response time. In the response to the RAIs, by letter dated November 14, 2017 (Reference 6), the licensee explained that the response time for all components will be listed in the TS Bases. The licensee provided markups of the TS Bases with the LAR. The NRC staff reviewed this component time response data to verify the proposed TS surveillance can be reasonably achieved by the licensee, and to confirm that the analytical limit for CREVS actuation could be satisfactorily maintained. However, the staff is not approving the TS Bases in this licensing action. For the AST model, the licensee assumed that the time for the CREVS actuation of the instrumentation to be ≤ 60 seconds. In the LAR, the licensee stated that this time was used in the different accident models as part of the time that it will take to switch from the normal operation mode to the emergency operation mode after the initial signal is sent.

As mentioned before, the licensee submitted an LAR to adopt AST on August 13, 2013 (Enclosure VI to Reference 41), which was later withdrawn. During the review of that amendment, the NRC staff issued several RAIs. Since these responses are relevant to this amendment, the licensee included them as Enclosure VII, "Responses to NRC RAIs on August 2013 Methodology Transition LAR Submittal (Non-Proprietary),"[8] of the letter dated January 17, 2017. The response to ARCB-RAI-13 explained how the total time to isolate the control room and switch to emergency mode operation was calculated. This response states that the time was calculated by establishing the time after generation of the isolation signal and the time for the control room ventilation isolation signal (CRVIS) to actuate the CREVS, plus additional margin.

In the response to the RAIs by letter dated November 14, 2017, the licensee provided additional details on how they estimated the response time. Specifically, the response time considers the response for each plant component necessary to actuate the CREVS, with the exception of the radiation detector, plus additional margin. The licensee used this estimated response time to model plant accidents in the dose analyses to adopt AST. At the frequency identified in SR 3.3.7.6, the licensee will measure the components response time, combine them and compare them to 60 seconds to ensure that the actual plant response is bounded by the response time modeled in the analyses.

In the same RAIs' responses, the licensee explained that the neutron radiation monitors are excluded from the ESF response time testing because the difficulty in generating an appropriate detector input signal and the principles of radiation detector operation ensure a virtually instantaneous response (in the order of microseconds), and therefore, negligible, when

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[8] Referred to as Enclosure VII, thereafter

compared to the 60-second criterion. As a consequence, the licensee added a Note to SR 3.3.7.6 stating that the radiation monitors are excluded from the ESF response time testing.

To support AST adoption, the licensee proposed a design modification to supply the CREVS control room isolation dampers with Class 1E battery power. By letter dated May 2, 2019 (Reference 86), the licensee confirmed that the proposed modification to supply the CREVS control room isolation dampers with Class 1E battery power has been completed.

In Enclosure IV (Reference 1), and the responses to RAIs by letter dated November 14, 2017 (Reference 6), the licensee explained that this modification will result in the system automatically isolating the CREVS by normal system operation when the control room air conditioning fan unit is turned off. There are two CREVS control room isolation dampers per safety train. The dampers are interlocked with the control room air conditioning unit and powered from the same safety-related source as the associated control room air conditioning unit. This source receives power from offsite sources if available, or an emergency diesel generator for a loss of offsite power (LOOP) scenario. The current operating procedure requires the operator to locally isolate the affected train of CREVS. With the proposed modification, the operator will be able to isolate the affected CREVS train from the control room when a failure occurs.

The NRC staff reviewed the new SR 3.3.7.6 and found that this was added to verify the actual response time and is bounded by the response time credited in the dose analyses. The staff found that the response time used in the dose analyses bounds the actual combined components response time. The staff also determined that the new SR is consistent with other SRs in the TS, and that the note to exclude the neutron detectors is due to the difficulty of testing the neutron radiation monitors. Therefore, the NRC staff concludes that the proposed SR 3.3.7.6 is acceptable and meets applicable criteria of 10 CFR 50.36(c)(3) to assure that the necessary quality and operability of the CREVS Actuation Instrumentation are maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

3.2 Human Factors Review

The NRC staff identified two areas of potential review in the submittal associated with human factors considerations as discussed below.

3.2.1 Maintenance of SG Water Level

Section 4.3.6.2.2.2, "Secondary Releases," of Enclosure IV discusses control rod ejection accident (Wolf Creek UFSAR Section 15.4.8.3) modeling assumptions that credit operator action to maintain SG water level. By letter dated January 15, 2018 (Reference 7), the licensee clarified that the function of the operator, in this circumstance, is to monitor rather than actively control the SG water level, and this action is not considered a time critical operator action.

During the actual event, SG water level will be automatically restored to the appropriate narrow range level when the SG low-level setpoint is reached. Loss of the minimum required SG water level would require multiple failures, which are beyond the design basis of the control rod ejection event. In addition, the Wolf Creek emergency operating procedures already contain direction for operators to continuously monitor that the SG water level is within the appropriate range on the narrow range scale. Given that operator action is not credited in the event design basis, and that the monitoring function is an existing emergency operating procedure-directed

operator activity, the NRC staff finds that no new operator actions are proposed that require human factors review.

3.2.2 Manual Isolation of Main Control Room Ventilation

Section 4.3.2.1, "Control Room Model," of Enclosure IV, states, in part:

> For events that rely solely on the control room air intake monitors for control room isolation, the unfiltered inleakage to the control room will continue to be associated with the normal mode air intake. This results in the modeling of the control room unfiltered inleakage with the normal mode atmospheric dispersion factors until either an SI [safety injection] actuation signal or manual operator action completes the control room isolation.

By letter dated January 15, 2018 (Reference 7), the licensee clarified that while the operators have the capability to manually complete the control room isolation during an event, no licensing basis analyses credit manual operator action to complete the control room isolation. Only two accident analyses consider manual action to isolate the control room: main steam line break (MSLB) analysis and the fuel handling accident (FHA) within containment with pathways to the auxiliary building open. In addition, the licensee also supplemented the original application by letter dated June 19, 2018 (Reference 10), to clarify that no change is being made to the current licensing basis overall containment isolation time of 60 seconds.

In the MSLB scenario, manual isolation of the control room is conservatively modeled to occur at the limiting time of the event. The FHA scenario includes a manual action to actuate a control room ventilation isolation signal with margin applied to the required action to enhance feasibility. The licensee stated that the subject FHA scenario is less limiting than the licensing basis scenario and was analyzed to respond to the NRC staff question during the acceptance review and would not be credited in the licensing basis. Given that no licensing basis analyses credit manual operator action to complete the control room isolation, the NRC staff concluded that no new operator actions are proposed that require human factors review.

However, in the supplement dated December 6, 2018 (Reference 14), page 10 of 19, the licensee revised the FHA analysis to credit the manual action to actuate a control room ventilation isolation signal within 30 minutes. On page 30 of 36 in Reference 7, the licensee provided a human factors analysis of the subsequently credited manual action to actuate isolation of the control room during an FHA. Therefore, the NRC staff evaluated the licensee's human factors analysis in accordance with the criteria provided in SRP Chapter 18, Attachment A, "Guidance for Evaluating Credited Manual Operator Actions" (Reference 18.k).

In Reference 7, the licensee stated that the control room operators would be quickly alerted to an FHA in containment due to the fact that the refueling team would be in constant communications with the control room during fuel handling operations. The control room operator response to manually initiate a CRVIS would then be procedurally directed. Control room operator response is assumed to occur within 10 minutes. The licensee notes that, unlike an accident that results in a reactor trip, no time is required to diagnose the event since the refueling team and control room are in constant communications. The licensee also notes in Reference 7 that the manual action to initiate a CRVIS is accomplished with a handswitch in the control room.

The licensee's human factors analysis in Reference 7 also states that the time credited in the FHA analysis for control room operators to manually initiate a CRVIS includes significant margin. Specifically, an additional 20 minutes has been added to the 10-minute manual action time assumed for the control room operators to initiate the CRVIS. This represents a margin of 200-percent allowed for the manual action.

The NRC staff finds that a CRVIS initiation by control room operators in response to an FHA in containment is an uncomplicated manual action consisting of manipulating a handswitch. Since the handswitch is located in the control room, no time is required for an operator to be dispatched and travel to a location remote from the control room. The NRC staff also finds that the 200-percent addition to the time credited in the FHA analysis to perform the manual action provides ample margin given that the action is procedurally directed, and event diagnosis is not required. While the manual action does not meet the licensee's criteria to be designated as a time critical operator action, normal procedure change controls associated with an FHA described in the licensee's UFSAR will provide an appropriate level of long-term integrity for the analysis-credited manual action.

Therefore, the NRC staff finds that the licensee's human factors analysis provided in its letter dated January 15, 2018, demonstrates that the ability of control room operators to manually initiate a CRVIS within 30 minutes of an FHA is feasible and reliable in accordance with the review criteria contained in Attachment A of SRP Chapter 18 (Reference 18.k).

3.3 Suppression Pool Elemental Iodine Evaluation – AST Adoption

According to NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Reference 42), iodine released from the damaged core to the containment after a LOCA is composed of 95 percent cesium iodide, which is a highly ionized salt, soluble in water. Iodine in this form does not present any radiological problems since it remains dissolved in the sump water and does not enter the containment atmosphere. However, in the radiation field existing in the containment, some of this iodine could be transformed from the ionic to the elemental form, which is scarcely soluble in water and can be, therefore, released to the containment atmosphere. Conversion of iodine to the elemental form depends on several parameters of which pH is very important. Maintaining pH basic in the sump water will ensure that this conversion will be minimized. The pH of the sump water in Wolf Creek is controlled by a sodium hydroxide (NaOH) buffer, which is formed by the addition of NaOH from the spray additive tank (SAT) and containment spray system to the boric acid (H_3BO_3) dissolved in the sump water after a LOCA. After a LOCA, several acids are either generated or are added to the containment. Relative amounts of these acids and that of NaOH determine the value of pH reached by the containment sump water.

To determine the sump pH in the calculation, the licensee derived and used molarities of NaOH and H_3BO_3 with verified pH titration curve data for aqueous solutions of NaOH and H_3BO_3.

After a LOCA, H_3BO_3 from the RCS, SI accumulators, RWST, and NaOH from the SAT are discharged into the sump. The licensee stated that two system alignments were analyzed in determining the sump pH. The two system alignments analyzed are the following: (1) two containment spray trains operating with one NaOH educator in service, and (2) two containment spray trains operating with two NaOH educators in service. The more limiting system alignment used in the analyses is the system alignment with two containment spray trains operating with one NaOH educator in service.

The Wolf Creek TS SAT minimum contained volume for NaOH solution is 4340 gallons. The licensee's calculation used a conservative delivered minimum volume of 2752 gallons of NaOH solution to determine the equilibrium pH post-LOCA. The parameters the licensee used in the sump pH analysis are summarized in Table 3.3-1 below.

Table 3.3-1. Design Parameters Used in Minimum Sump pH Analyses

Component	Volume or Mass	Concentration
RWST	379,800 gallons	2500 parts per million (ppm) H_3BO_3
RCS	504,520 pounds-mass (lb_m)	1980 ppm H_3BO_3
SI Accumulators	26,376 gallons	2500 ppm H_3BO_3
SAT	2752 gallons	28 weight percent NaOH
Electrical Cables	50,000 lb_m	17.5 weight percent Chlorine (Cl)*

*Value obtained from methodology provided in NUREG/CR-5950, "Iodine Evolution and pH Control," (Reference 43).

Based on the information provided in Table 3.3-1 above, the NRC staff calculated the mass quantities provided in Table 3.3-2 below.

Table 3.3-2. Staff Derived Mass Quantities Based on Design Parameters

Component	Compound	Pound-mass
RWST	H_3BO_3	7933
RCS	H_3BO_3	999
SI Accumulators	H_3BO_3	551
SAT	NaOH	8370*
Electrical Cables	Hydrochloric acid	1.86**

*Based on density of 10.93 lb_m per gallon of NaOH aqueous solution
**pound-mass per Megarad, value derived from methodology provided in NUREG/CR-5950.

The licensee determined the maximum sump volume to be 1.87×10^{06} liters. Using the maximum sump volume and the information provided in the tables above, the NRC staff determined the total mass for H_3BO_3 to be approximately 9483 (lb_m) or 3.7×10^{-02} gram-moles (g-moles) per liter.

In addition, the licensee considered the effects of strong acid generation on the post-LOCA sump pH. Based on the guidance in NUREG/CR-5950, the licensee calculated the mass of nitric acid (HNO_3) generated by the radiolysis of air and water inside containment and the mass of hydrochloric acid (HCl) generated by the radiolysis of Hypalon® electrical cable insulation inside containment.

Hydrochloric acid is formed from decomposition of chlorinated polymer cable insulation by radiation. The licensee estimated the cable insulation to weigh approximately 50,000 lb_m. Based on this weight and the guidance in NUREG/CR-5950, the amount of HCl produced by the irradiation of electrical cable is estimated as 4.6×10^{-04} g-moles of HCl per lb_m of insulation per Megarad (Mrad). The NRC staff determined that the total amount of HCl produced per Mrad, based on the mass of cable insulation determined by the licensee, is 1.23×10^{-05} g-moles per liter per Mrad. The amount of HNO_3 produced is proportional to the time-integrated dose rate for gamma and beta radiation. Although the licensee did not report the amounts of HCl and HNO_3 used in the calculation, the NRC staff calculated these values and compared the results to calculations from similar plants. Comparison of the parameters affecting generation of strong

acids in the post-LOCA environment at Wolf Creek and in similar plants has shown very close similarity. Both acids are strong acids and will contribute to lowering the sump pH. With the amount of strong acids generated in the containment, in 30 days after a LOCA in similar plants, the decrease of sump water pH was less than 0.1 units. As such, the staff has determined that Wolf Creek will have a similar decrease in sump pH due to strong acid generation. The licensee reported that the moles of strong acids generated inside containment post-LOCA were assumed to instantaneously be neutralized by a molar equivalent of NaOH, which in turn, reduced the amount of NaOH that was available to neutralize the H_3BO_3 injected from the RCS, SI accumulators, and RWST.

In order to neutralize the H_3BO_3, HCl and HNO_3, the licensee chose to buffer the sump pool water by using a NaOH buffer. Such buffering action is intended to maintain basic pH in the sump pool despite the presence of the acids. The licensee has calculated that by adding approximately 8370 lb_m or 5.1×10^{-02} g-moles per liter of NaOH, at a reference temperature of 77 °F and density of 10.86 lb_m per gallon (based on "Perry's Chemical Engineering Handbook") from the SAT to the sump pool during the injection phase, it will maintain the pH in the sump water as basic.

Based on the results of the analysis, the licensee determined that the sump pH, with the more limiting system alignment, results in a pH of 7 in approximately 11 minutes, with a final long-term pH of 8.7 in approximately 70 minutes, under post-accident conditions. With the less limiting system alignment (i.e., two NaOH educators in service) the sump pH was determined to be 7 in approximately 9 minutes, with the final long-term pH of 8.7 in approximately 35 minutes.

Based on the above and the information provided by the licensee, the NRC staff concludes that by using NaOH as a buffer in the quantity specified, the pH of the sump will remain above a pH of 7 for 30 days post-LOCA.

3.4 Atmospheric Dispersion Factors

3.4.1 Atmospheric Dispersion Estimates

The licensee's request for this LAR uses a new set of atmospheric dispersion factors (also known as χ/Q values and relative concentrations) for the exclusion area boundary (EAB), the outer boundary of the low population zone (LPZ), the control room, and the technical support center (TSC) receptors in the calculations made for the radiological consequence assessments. The NRC staff reviewed the licensee's new atmospheric dispersion analyses as described below.

3.4.2 Meteorological Data

In support of the atmospheric dispersion analysis presented in Enclosure V, "Meteorological Data Used to Determine Offsite Control Room and TSC Atmospheric Dispersion Factors," of the letter dated January 17, 2017 (Reference 1), the licensee provided hourly onsite meteorological data from January 1, 2006, through December 31, 2010. The meteorological data were formatted for the ARCON96 atmospheric dispersion code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes") (Reference 44) to calculate updated χ/Q values for the control room and TSC. This format contained hourly data on wind speed, wind direction, and atmospheric stability class taken from the 10-meter (m) and 60-m levels of the onsite meteorological tower.

The NRC staff has completed a detailed review related to the acceptability and representativeness of the hourly meteorological data using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data" (Reference 45). Examination of the data revealed generally stable and neutral atmospheric conditions at night and unstable and neutral conditions during the day, as expected. Wind speed, wind direction, and stability class frequency distributions for each measurement channel were similar from year-to-year. Based on this review, the NRC staff considers the onsite meteorological database suitable for use in making calculations for the atmospheric dispersion analyses used to support this LAR.

The licensee also provided a wind speed, wind direction, and atmospheric stability joint frequency distribution (JFD) for the same 5 years as an input to the PAVAN atmospheric dispersion computer code (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," dated September 1982 (Reference 46), to calculate the updated χ/Q values for the EAB and outer boundary of the LPZ. The wind data were obtained from the 10-m level of the onsite meteorological tower, and the stability data were derived from the vertical temperature difference (delta-temperature) measurements taken between the 60-m and 10-m levels on the onsite meteorological tower in accordance with RG 1.23 (Reference 26). The NRC staff developed its own JFD from the hourly data provided by the licensee and determined the NRC staff's JFD to be commensurate with the licensee's JFD.

3.4.3 Onsite Control Room Atmospheric Dispersion Estimates

In support of the LAR, the licensee used the computer code ARCON96 to estimate χ/Q values for the control room and TSC for potential accidental releases of radioactive material. RG 1.194 (Reference 30) endorses the ARCON96 model for determining χ/Q values to be used in the design basis evaluations of control room radiological habitability.

The ARCON96 code estimates χ/Q values for various time-averaged periods ranging from 2 hours to 30 days. The meteorological input to ARCON96 consists of hourly values of wind speed, wind direction, and atmospheric stability class. The χ/Q values calculated through ARCON96 are based on the theoretical assumption that material released to the atmosphere will be normally distributed (Gaussian) about the plume centerline. A straight-line trajectory is assumed between the release points and receptors. The diffusion coefficients account for enhanced dispersion under low wind speed conditions and in building wakes.

The hourly meteorological data are used to calculate hourly relative concentrations. The hourly relative concentrations are then combined to estimate concentrations ranging in duration from 2 hours to 30 days. Cumulative frequency distributions, prepared from the average relative concentrations and the relative concentrations that are exceeded no more than 5 percent of the time for each averaging period, are determined.

The dispersion coefficients used in ARCON96 have three components. The first component is the diffusion coefficient used in other NRC models such as PAVAN. The other two components are corrections to account for enhanced dispersion under low wind speed conditions and in building wakes. These components are based on analysis of diffusion data collected in various building wake diffusion experiments under a wide range of meteorological conditions. Because the dispersion occurs at short distances within the plant's building complex, the ARCON96 dispersion parameters are not affected by nearby topographic features such as bodies of water.

Therefore, the NRC staff concludes that the licensee's use of the ARCON96 dispersion parameter assumptions acceptable.

The licensee provided the following as the necessary input to ARCON96:

- Onsite Hourly Meteorological Data: 2006 through 2010
- The following Tables from Enclosure IV (Reference 1):
    - Table 4.1.2-1: Input Parameters for Emergency Control Room Air Intake
    - Table 4.1.2-2: Input Parameters for TSC Air Intake
    - Figure 4.1.2-1: Diagram of Source and Receptor Locations for [Wolf Creek]

Three receptor (i.e., air intake) points, the Emergency Control Room Intake, Normal Control Room Intake, and TSC Air Intake, were modeled for the following seven release points:

- Reactor Building Equipment Hatch
- Unit Vent Stack
- MSSVs/Atmospheric Relief Valves (MSSV/ARV) Vent
- RWST Vent
- Turbine-Driven Auxiliary Feedwater Exhaust Vent
- Radwaste Building (Nearest Point to Receptor)
- Reactor Building Wall (Nearest Point to Receptor)

The NRC staff confirmed the licensee's atmospheric dispersion estimates by running the ARCON96 computer model and obtaining similar results (i.e., values on average within ± 0.11 percent). Both the NRC staff and licensee used a ground-level release assumption for each of the release/receptor combinations as well as the source-receptor distances and directions provided in Enclosure IV, Tables 4.1.2-1(a), 4.1.2-1(b), and 4.1.2-2. Based on the results of its confirmatory analysis, the staff concludes that the licensee's control room and TSC χ/Q values are acceptable.

The NRC staff issued an RAI by e-mail dated June 14, 2017 (Reference 47), requesting that the licensee provide clarification on potential conflicts in release heights reported in Enclosure IV, Tables 4.1.2-1(a), 4.1.2-1(b), and 4.1.2-2. In its response by letter dated July 13, 2017 (Reference 4), the licensee provided a complete explanation for the apparent differences in source release heights provided in the aforementioned tables. The licensee's letter stated the values in the tables are supported by the assumptions provided in Section 4.1.2.3, "Assumptions and Acceptance Criteria," of Enclosure IV. The NRC staff reviewed the information provided in Section 4.1.2.3 and finds that the licensee's response is acceptable and resolved the NRC staff's concern.

3.4.4 Offsite EAB and LPZ Atmospheric Dispersion Estimates

In support of the LAR, the licensee used the computer code PAVAN to estimate χ/Q values at the EAB and outer boundary of the LPZ for potential accidental releases of radioactive material. PAVAN implements the methodology described in RG 1.145 (Reference 29) for determining χ/Q values at the EAB and LPZ outer boundary.

The licensee calculated EAB and LPZ outer boundary χ/Q values using meteorological data from January 1, 2006, through December 31, 2010. The licensee chose to implement the

diffusion parameter assumptions outlined in RG 1.145 as a function of atmospheric stability for its PAVAN model runs. The NRC staff evaluated the applicability of the PAVAN diffusion parameters and concluded that no unique topographic features (such as rough terrain, restricted flow conditions, or coastal or desert areas) preclude the use of the PAVAN model for the Wolf Creek site. The licensee applied terrain correction factors "to account for the possibility of non-straight trajectory due to temporal and spatial variations in airflow in the EAB and LPZ areas that do not reflect in the meteorological data collected at a single onsite station." The NRC staff recognizes this to be a conservative assumption, and therefore, finds that the licensee's use of the PAVAN diffusion parameters, as outlined in RG 1.145, with terrain correction factors applied, acceptable.

The licensee modeled two ground-level release points: one for releases near containment and a second for releases from the RWST. The cross-sectional area of the containment building was used to determine χ/Q values for releases near the containment whereas the cross-sectional area of the RWST was used for releases from the RWST. The NRC staff confirmed that including the building wake effects for a ground-level release has minor influence on the predicted χ/Q values. A ground-level release assumption that assumes the appropriate building dimensions is acceptable to the staff because the PAVAN model includes both plume meander and building wake effects, which are mutually exclusive.

Using the information provided by the licensee, including the 10-m level JFDs of wind speed, wind direction, and atmospheric stability presented in Enclosure IV, Tables 4.1.1-11 through 4.1.1-17, the NRC staff confirmed the licensee's χ/Q values by running the PAVAN computer code and obtaining consistent results. On the basis of this review and the NRC staff's confirmatory calculations using PAVAN, the NRC staff concluded that the EAB and LPZ outer boundary χ/Q values were acceptable for use in the proposed radiological consequence assessments in support of this LAR for adoption of AST.

3.4.5 Summary

The NRC staff reviewed the methodology used by the licensee to derive the χ/Q values associated with postulated releases from potential release points. The NRC staff performed an extensive screening of meteorological data and found the licensee used appropriate atmospheric dispersion models to derive the resulting χ/Q values in accordance with NRC staff regulatory guidance in SRP, Sections 2.3.4, and 6.4; RGs 1.23, 1,145, and 1.194; and meets the requirements of GDC 19 of Appendix A to 10 CFR Part 50 with respect to the meteorological considerations used to evaluate the personnel exposures inside the control room during radiological airborne hazardous material accident conditions. On the basis of this review and the staff's confirmatory atmospheric dispersion analyses, the NRC staff concluded that the meteorological data and the resulting onsite and offsite χ/Q values are acceptable for use in the radiological consequence assessments supporting this LAR for adoption of the AST.

3.5 Adoption of Alternative Source Term

The licensee performed analyses for the full implementation of the AST using the guidance in RG 1.183 (Reference 17) and SRP, Section 15.0.1 (Reference 18.h). In its LAR, the licensee stated that the AST methodology is being used to make the following changes:

- The control room habitability envelope unfiltered inleakage will be revised from 20 standard cubic feet per minute (scfm) to 50 scfm.

- The control building unfiltered inleakage will be revised from 300 scfm to 400 scfm.
- Wolf Creek UFSAR Chapter 15 dose analyses will be revised in accordance with the guidance in RG 1.183.
- The TSs will be revised to address the update of the accident source term and associated DBAs utilizing the guidance provided in RG 1.183 and the associated control room dose limits of GDC 19 of Appendix A to 10 CFR Part 50, and offsite dose limits of 10 CFR 50.67.

The NRC staff reviewed the technical analyses related to the radiological consequences of DBAs that were performed by the licensee in support of this proposed license amendment. Information regarding these analyses was provided in Enclosure IV, as supplemented by letters dated May 4, 2017, January 15, January 29, April 19, June 19, and December 6, 2018 (References 3, 7, 8, 9, 10, and 14, respectively). The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess these impacts and performed independent calculations to confirm the conservatism of the WCNOC analyses. The NRC staff also considered relevant information in the Wolf Creek UFSAR (Reference 20) and the TSs. The NRC staff's technical evaluation is provided below.

RAIs were developed during the NRC staff's review. Licensee responses to those RAIs are summarized and referenced, where appropriate, throughout this section. The following paragraphs provide a brief description of the systems supporting the proposed TS changes related to the adoption of the AST.

Control Room Emergency Ventilation System (CREVS)

The CREVS provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity. The CREVS consists of two independent, redundant ventilation trains that recirculate, cool, pressurize and filter the air in the control room boundary.

The control room boundary contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. The area is protected during normal operation, natural events, and accident conditions. The boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the control room envelope. The OPERABILITY of the boundary must be maintained to ensure that the inleakage of unfiltered air into the control room will not exceed the inleakage assumed in the licensing basis analysis of DBA consequences to control room occupants.

The CREVS is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation. Upon receipt of the actuating signal(s), normal air supply to the control room is isolated, and the system is aligned for recirculation. A portion of the recirculation air flows through the filters. Pressurization and filtration of the air supply is also initiated in the emergency mode of operation.

The emergency mode of operation closes the unfiltered outside air intake and unfiltered exhaust dampers and aligns the system for recirculation. A portion of the recirculation control room air flow through the redundant filtration system trains of high-efficiency particulate air filters and the charcoal adsorbers.

CREVS Actuation Instrumentation

The CREVS Actuation Instrumentation consists of two radiation monitors in the control room air intake and four radiation monitors in the containment purge isolation system. A high radiation signal from any of these gaseous detectors, called a CRVIS, will initiate both trains of the CREVS and place the system into the emergency mode of operation. The CRVIS also initiates pressurization and filtered ventilation of the air supply to the control room.

RCS Specific Activity

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of an accident.

For the revised accident source term, the maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 50.67. The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 50.67 limits during analyzed transients and accidents.

The LCO contains specific activity limits for both Dose Equivalent (DE) I-131 and DE XE-133 gross specific activity. The allowable levels are intended to limit the 2-hour dose at the site boundary to a small fraction of the 10 CFR 50.67 dose guideline limits.

Containment Penetrations

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident such that offsite radiation exposures are maintained within the requirements of 10 CFR Part 100.

During core alterations or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment with containment penetration closure.

Emergency Exhaust System (EES)

The fuel building ventilation system consists of the fuel building supply system, the normal exhaust system, and the EES. The EES serves both the auxiliary building and the fuel building.

In the event of an FHA, the EES collects and processes the airborne particulates in the fuel building. In the event of a LOCA, the EES processes the atmosphere of the auxiliary building.

The EES serves the auxiliary building only following a LOCA to assure that all ECCS leakage to the auxiliary building atmosphere; and the containment air purged via the hydrogen purge system are processed.

The system has two independent and redundant trains, each consisting of a heater, prefilter, a high-efficiency particulate air filter, charcoal absorber and fan. Portions of the system are in operation during normal operation. The system is automatically started by either a SI signal (indicative of a potential LOCA) or a fuel building isolation signal (indicative of a potential fuel handling accident). In the LOCA mode, the system is aligned to the auxiliary building exhaust. In the fuel building isolation mode, the fuel building is isolated.

3.5.1 Fission Product Inventory and Release Fractions

3.5.1.1 Fission Product Inventory

RG 1.183, Regulatory Position 3.1, "Fission Product Inventory", states, in part:

> The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2... or ORIGEN-ARP....

The source term information for use in the radiological consequence analyses was generated using the ORIGEN-S ("SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," ORNL/TM-2005/39, Volume II, Book 1, dated February 2005). According to Enclosure IV, Section 5, Table A, "Conformance with RG 1.183 Main Sections," the WCNOC's analyses conforms to RG 1.183, Regulatory Position 3.1 and the following comments:

> The inventory of fission products in the reactor core and available for release to the containment was based on the maximum full power operation with a core thermal power of 3637 MWt [megawatt thermal] (102% of 3565 MWt nominal power).
>
> Core design parameters (enrichment, burnup, and MTU [metric ton uranium] loading) are based on the cycle 19 core design. Margin[9] is added to the EOC [end of cycle] core inventory, calculated with ORIGEN-S, to account for potential core design differences in future cycles. The magnitude of this margin is based on sensitivity studies that consider variations in enrichment and burnup.

The licensee assumed a core with seven fuel regions with a maximum assumed enrichment for a single fuel assembly.[10] The length of the irradiation for the cycle modeled (Cycle 19) used to determine the end of life radionuclide values is 511 effective full power days. The licensee states and the NRC staff agrees that using these burnups for the prior and current cycles should be sufficient in duration to allow the activity of the dose-significant radionuclides to reach equilibrium or maximum values.

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[9] The margin added is provided in the licensee's response to RAI ARCB-RAI-2 in Enclosure VI to letter dated January 17, 2017. The value is considered proprietary.

[10] The maximum assumed enrichment (weight percent Uranium (U) -235) for a single assembly is provided in the licensee's response to RAI ARCB-RAI-4 in Enclosure VI to the letter dated January 17, 2017. The value is considered proprietary.

3.5.1.2 Fission Product Release Fractions

As stated in RG 1.183, Regulatory Position 3.2, "Release Fraction," the release fractions associated with the LWR core inventory released into containment for the DBA LOCA and non-LOCA events have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 MWD/MTU. For non-LOCA events, the acceptability of the RG 1.183, Table 3, "Non-LOCA Fraction of Fission Product Inventory in Gap," release fractions are further restricted for use with fuel with a maximum linear heat generation rate less than 6.3 kw/ft peak rod average for burnups exceeding 54,000 MWD/MTU. Table A of Enclosure IV states that the Wolf Creek analysis conforms to Regulatory Position 3.2. Based upon the licensee's statements regarding their conformance to Regulatory Position 3.2 and the NRC staff's review of the gap fractions discussed in Section 3.6.7.1 of this SE, the NRC staff finds the gap fractions used in the radiological analysis is acceptable.

3.5.2 Reactor Coolant and Secondary Plant Radiation Source Term

The initial RCS activity for the accident analyses are based on operation with 1 percent fuel defects, except for the initial iodine and xenon. The initial RCS iodine, xenon, and krypton (Kr) were scaled down to the TS LCO 3.4.16, "RCS Specific Activity," limit of less than or equal to 1.0 microcurie per gram of DE I-131 and 500 microcuries per gram of DE XE-133. Consistent with RG 1.183, Regulatory Position 3.1, the coolant activities are based on a core power of 3,637 MWt, which is the nominal core power of 3,565 MWt plus a 2 percent calorimetric uncertainty.

The FIPCO-VI computer code ("FIPCO-VI A Computer Code for Calculating the Distribution of Fission Products in Reactor Systems," dated March 2004) was used by the licensee to calculate the buildup of fission product activities in plant systems and components, including the RCS, chemical and volume control system (CVCS) demineralizer resins, volume control tank (VCT) liquid and vapor phases, and waste gas decay tank (WGDT). The time-dependent inventory of the core fission products calculated by ORIGEN-S is used as input to the FIPCO-VI evaluations. When calculating the radionuclide inventory in the VCT, no purging is assumed through the cycle. The activities for the WGDT are calculated assuming a maximum RCS letdown rate to degas the RCS by repeated purges of the VCT at EOC. The fuel management multiplier is applied to all results calculated by FIPCO-VI. The licensee also stated that that the maximum values calculated for each radionuclide throughout the operating cycle results were determined for the reactor coolant specific activity VCT liquid and vapor phase specific activity and the WGDT activities.

The use of ORIGEN is in accordance with RG 1.183 guidance and is therefore, acceptable to the NRC staff. The source terms calculated for WCNOC by the FIPCO code were reviewed, and when possible, compared to any previously calculated source terms. Based upon this comparison of source terms and the use of the maximum values calculated for each radionuclide throughout the operating cycle, the NRC staff finds these proposed source terms reasonable and acceptable for the WCNOC DBA analyses reviewed by the NRC staff in this SE.

In the licensee's response to RAI ARCB-RAI-27 in Enclosure VII of letter dated January 17, 2017 (Reference 1), the licensee stated that a spike in cesium (Cs) associated with the iodine spikes (both pre-accident and accident-initiated) in the Wolf Creek radiological consequences analyses MSLB, SG tube rupture (SGTR), loss of non-emergency alternating current (LOAC) power, and letdown line break (LLB) was not explicitly modeled, however, the licensee stated that the initial Cs activity modeled in the RCS bounds the Cs, which would be released to the

RCS in association with iodine spiking. Based upon the licensee's response to RAI ARCB-RAI-27 in Enclosure VII, and that the Cs activity initially modeled in the RCS is based upon the design basis fuel defect level of 1 percent, the NRC staff agrees that the initial cesium activity modeled in the RCS bounds the cesium that would be released to the RCS associated with iodine spiking. Therefore, the Cs activities assumed in the Wolf Creek analyses for the MSLB, SGTR, LOAC, and LLB are acceptable to the NRC staff.

3.5.3 Dose Conversion Factors

The TEDE dose is the sum of the committed effective dose equivalent from internal exposure and the effective dose equivalent (EDE) from external exposure. The EDE is used in lieu of the deep dose equivalent in determining the contribution of external dose to the TEDE, consistent with the guidance provided in RG 1.183. The dose conversion factors (DCFs) used in determining the committed effective dose equivalent dose are from Table 2.1 of Federal Guidance Report No. 11 (Reference 48), and the DCFs used in determining the EDE dose are from Table III.1 of Federal Guidance Report No. 12 (Reference 49). The licensee used the appropriate committed EDE and EDE DCFs from Federal Guidance Reports 11 and 12 to determine the TEDE dose in accordance with AST evaluations.[11] The use of DCFs from Federal Guidance Report 11 and Federal Guidance Report 12 is in accordance with RG 1.183, Regulatory Positions 4.1.2, 4.1.4, and 4.2.7 and the proposed TS definitions of DE-I-131 and DE-XE-133 that specify these DCFs are consistent with the licensee's evaluations and analyses used to determine the radiological consequences for the applicable DBAs in the UFSAR (Reference 20) and therefore, are acceptable to the NRC staff.

3.5.4 Control Room and TSC

At the start of all the events considered, the control room ventilation system is in normal mode. In this mode, unfiltered air from the environment enters the control building and control room. Receipt of a SI actuation signal or a high radiation signal from the control room air intake monitors will isolate the control room and initiate the emergency mode of operation, including a delay.

After emergency mode is initiated, outside air is brought into the control building through safety grade filters. Makeup air is brought into the control room via both trains of the control room filtration system, which draws in air from the control building. Unfiltered air also leaks into the control building and control room via assumed inleakage rates. In addition, a filtered recirculation flow is modeled for the control room during emergency mode operation. Air in the control room and control building is discharged at flow rates to match the total inflow to the compartments.

The control room ventilation isolation signal starts both trains of the control room filtration system. However, a failure of one of the filtration fans is assumed at the start of emergency mode and a larger unfiltered inflow to the control room is assumed since only half of the makeup flow to the control room can pass through a filter. After a defined time of 90 minutes, operator

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[11] Per the WCNOC response to RAI ARCB1-General-1 in letter dated January 15, 2018, the dose conversion factor for Cesium (Cs)-137 has a very low EDE value of 7.74E-18 Sv-m^3/Becquerel-second. However, its short-lived daughter nuclide Barium (Ba)-137 m has a large EDE of 2.88E-14 Sv-m^3/Becquerel-second. The daughter value was substituted for the parent nuclide value in the WCNOC analyses.

action isolates the failed train and reduces the unfiltered inflow to the control room, and consequently lowers the filtered inflow to the control building.

In Enclosure IV, Table 4.3-5, "Control Room and Control Building Parameters," the licensee provided the parameters modeled in the control room personnel dose calculations and those in the current licensing basis. Some of these inputs and parameters are modified by RAI responses to RAI ARCB1-GENERAL-2 and ARCB1-GENERAL-3, by letters dated June 19, 2018 (Reference 10), and December 6, 2018 (Reference 14). In the response to RAI ARCB1-GENERAL-2 by letter dated June 19, 2018, the licensee stated that the normal mode control building and control room flows were increased by 10 percent and rounded up in value to account uncertainty listed in Section 5.5.11, "Ventilation Filter Testing Program (VFTP)," of the Wolf Creek TSs. Increasing these normal flow rates to account for uncertainty is consistent with RG 1.183, Regulatory Position 5.1.3, "Assignment of Numeric Input Values," which states, in part that "[i]f a range of values or a tolerance band is specified, the value that would result in a conservative dose should be used" and therefore, is acceptable to the NRC staff. The control room and control building ventilations flow paths and flows, and the control room modeling assumptions for the timing for the high radiation signal, SI signal, emergency mode actuation and normal intake damper closure are provided in response to RAI ARCB1-GENERAL-3 by letter dated June 19, 2018 (e.g. Table 3, "Control Room Modeling Assumptions," Table 4 , "Control Room and Control Building Ventilation Flows" and Figure 3, "Control Room and Control Building Ventilation Flows"), and by letter dated December 6, 2018.

In the supplemental responses to RAI ARCB1-GENERAL-3, by letters dated June 19, 2018, and December 6, 2018, the licensee assumed that the total unfiltered inleakage modeled during emergency mode is 50 cubic feet per meter (cfm) for the control room. Prior to the closure of the normal heating, ventilation, and air conditioning (HVAC) intake damper, the 50-cfm unfiltered inleakage is associated with the normal HVAC intake atmospheric dispersion factor. After closure of the normal HVAC intake damper, the unfiltered inleakage is apportioned between the emergency mode HVAC intake (40 cfm) and the communications corridor intake (10 cfm associated with ingress/egress).

The NRC staff reviewed the assumption of 10 cfm unfiltered inleakage for ingress and egress. Chapter 3 of the UFSAR, Appendix 3A (Reference 20), discusses the extent to which Wolf Creek conforms to NRC published regulatory guides. No exception to RG 1.197, Regulatory Position 2.5, "Inleakage Test Acceptance Criteria," is noted. Regulatory Position 2.5 states, in part:

> Any analysis to demonstrate that a facility meets GDC-19 should include a value for inleakage that is due to ingress to and egress from the CRE. This value is combined with the baseline test value for inleakage in the analyses. When integrity tests are performed to determine the CRE's integrity characteristics, the acceptance criterion for the test should be the licensing basis amount less the amount designated for ingress and egress. The staff considers 10 cfm as a reasonable estimate for ingress and egress for control rooms without vestibules.

The NRC staff finds that the assumption of 10-cfm unfiltered inleakage for ingress and egress is consistent with RG 1.197, Regulatory Position 2.5, and therefore, is acceptable, except for, in one portion of one design basis analysis. The value of 10 cfm accounts for only the sweeping action of doors as they are opened or closed. The design-basis analysis described in Section 3.5.6.2.6, "FHA in an Open Containment and Release to Control Room via the Auxiliary Building," of this SE, uses a larger value for a portion of the analysis to account for inflow

through open doors due to pressure across the door boundary that is not associated with the sweeping action of the doors.

In Enclosure IV, Table 4.3-16, the licensee provided the parameters modeled in the proposed TSC personnel dose calculations and those in the current licensing basis. Based upon Table 4.3-16, "Technical Support Center Parameters," the TSC volume, the TSC filter efficiencies, and time of the delay to switch to the emergency mode of operation after event initiation were all changed from the current licensing basis. At the start of all the events considered, the TSC ventilation system is in normal mode. In this mode, unfiltered air from the environment enters the TSC. After emergency mode is manually initiated, outside makeup air is brought into the TSC through safety grade filters. Unfiltered air also leaks into the TSC via an assumed inleakage rate. In addition, a filtered recirculation flow is modeled during emergency mode operation. Air is discharged at a flow rate to match the total inflow to the compartment.

The changes in the control room and TSC models were incorporated into the NRC staff's independent assessment of the DBA consequence used to assess control room habitability and TSC doses. The results of these analyses are discussed below.

The acceptance criteria for the control room is taken from 10 CFR 50.67 and is 5 rem TEDE. Based on the guidance provided in NUREG-0696, Section 2.6, "Habitability" (Reference 33), the radiological habitability for the TSC is to be the same as the control room under accident conditions, so the acceptance criteria for the TSC is also 5 rem TEDE. Both the control room and the TSC doses are evaluated for the duration of the accident, which is assumed to be 30 days.

3.5.4.1 Control Room Operator Dose during Ingress and Egress

Section 50.67(b)(2) of 10 CFR requires that the licensee's analysis demonstrates with reasonable assurance that adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. This section of the SE discusses the NRC staff's review of the licensee's evaluation of the control room operator dose during access to the control room.

The licensee evaluated the dose received by the control room operators during access to the control room for the 30-day period following the LOCA. The assumption inherent in the transit dose is that the operator travels to the site, parks in the parking lot, and walks from the parking lot to the control building via the turbine building entrance without any supplemental radiation protection (e.g. decontamination, protective equipment or prophylactic drugs). The operator is assumed to leave using the same path.

The transit dose scenario modeling and assumptions used are generally consistent with the design-basis LOCA offsite dose assumptions. A summary of the assumptions and inputs are provided in the supplemental response to RAI ARCB1-CONTROL ROOM-3, by letter dated January 15, 2018 (Reference 7) (except for the operator breathing rate, which is provided in a supplement to RAI ARCB1-CONTROL ROOM-3 dated June 19, 2018 (Reference 10), and further descriptions and justification for the deposition factors assumed for the outside environment that are provided in a supplement to RAI ARCB1-CONTROL ROOM-3 by letter dated December 6, 2018 (Reference 14)). The NRC staff's evaluations of the assumed breathing and outside deposition rates are provided below.

In the supplemental response to RAI ARCB1-CONTROL ROOM-3, by letter dated June 19, 2018, the operator breathing rate assumed for operator transit to and from the control room is 7.0E-04 cubic meters/second, which is twice the breathing rate assumed in RG 1.183, Regulatory Position 4.2.6, for modeling the dose to operators within the control room. The NRC staff finds that the proposed value for the breathing rate is acceptable because: (1) it bounds the largest 95th percentile breathing rate for moderate intensity activities contained in Table 6-2 of U.S. Environmental Protection Agency's (EPA's): Exposure Factors Handbook for Recommended Short-Term Exposure Values for Inhalation (Males and Females Combined) (Reference 50)[12] and (2) the NRC staff considers the transit by the operator to be a moderate intensity activity given the assumed rate of speed and the distances traveled by the operator.

The NRC staff reviewed the justification of the assumed deposition rates provided in Inputs/Assumption No. 10 of the supplemental response to RAI ARCB1-CONTROL ROOM-3, by letter dated January 15, 2018, and further discussed in a supplement to RAI ARCB1-CONTROL ROOM-3, by letter dated December 6, 2018. The NRC staff finds that the assumption that noble gases are not assumed to deposit on the ground to be reasonable and consistent with the treatment of noble gases in RG 1.183, and is therefore, acceptable. However, the NRC staff does not find the assumption for the deposition for particulates and iodine acceptable based upon the justification provided. The uncertainty in these values due to environmental conditions and the use of short term values (on the order of 1 hour) for an accident that is assumed to be 30 days has not been adequately justified. The NRC staff based its acceptance of the transit dose for the LOCA on the licensee's dose results, as described below.

In the supplemental response to RAI ARCB1-CONTROL ROOM-3, by letter dated June 19, 2018, the licensee stated the transit dose to the operator from ingress and egress is 0.8 rem. The licensee assumed that contributions to the dose occur from containment leakage, engineered safety features system leakage, RWST back leakage, direct dose from ground shine from deposited radioactivity, and direct containment shine. The NRC staff finds the transit dose to the operator from ingress and egress to be acceptable because: (1) the proposed dose is reasonable based upon the NRC staff's previous experience reviewing the ingress and egress operator dose for other licensees, (2) unless otherwise noted above, the licensee used conservative assumptions consistent with, and where appropriate with, the LOCA analysis for offsite and control room doses, (3) all known pathways to the environment that could contribute to the operator dose are considered, and (4) no supplemental radiation protection (e.g. decontamination, protective equipment or prophylactic drugs) is credited in the analysis. Therefore, the NRC staff finds the dose results for the control room operator transit dose during ingress and egress to be acceptable for the AST LOCA analysis.

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[12] The largest moderate intensity activity level breathing rate in Table 6-2 of the EPA Exposure Factors Handbook is 4.0E-02 cubic meters/minute or 6.7E-04 cubic meters/second which is bounded by the assumed value of 7.0E-04 cubic meters/second.

3.5.4.2 Control Room Operator Direct Shine Evaluations

RG 1.183, Regulatory Position 4.2.1 states:

> The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include:
>
> - Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility,
> - Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope,
> - Radiation shine from the external radioactive plume released from the facility,
> - Radiation shine from radioactive material in the reactor containment,
> - Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.

Enclosure IV, Table A, states that the Wolf Creek limiting control room dose analysis included the above items listed in Regulatory Position 4.2.1, and that the dose analysis conforms to this regulatory position. The licensee stated that the analysis considered all significant sources of radiation that would cause exposure to control room personnel. In a supplemental response to ARCB1-GENERAL-3, dated June 19, 2018, the calculated dose to control room personnel from external sources from a DBA LOCA was calculated to be 0.126 rem TEDE.

In its supplemental response to RAI ARCB1-CONTROL ROOM-1 dated January 15, 2018, the licensee stated that the shine dose contribution to the post-LOCA control room dose from activity in the control building was calculated using control building radionuclide inventories, and that the activity in the control building was assumed to be uniformly distributed throughout the control building, excluding the control room. The licensee stated that the shine dose calculations were performed using the SCAP-II computer code[13]. In WCNOC's supplemental response to RAI ARCB1-CONTROL ROOM-1 dated December 6, 2018, the licensee stated that, "[t]he post-LOCA 30-day integrated dose to the control room operators from activity dispersed within the control building was calculated to be approximately 26.2 mrem [millirem]...," and that this activity is not a significant dose contributor. Furthermore, the licensee stated, in part, that "the activity accumulated the control building [for events other than the LOCA] would be bounded by that of the LOCA" and that "the [dose] contribution to the LOCA of 26.2 millirem would be bounding for the events other than the LOCA." The licensee also stated, in part, that "this [dose] is within the rounding applied to events other than the

[13] The SCAP-II computer code is a point kernel program employing dependent point kernel or single/albedo scatter methods to calculate the radiation level at a detector point located within or outside a complex scattering geometry describable by the combinatorial geometry method.

LOCA" and that "the contribution to control room doses for events other than the LOCA was considered and judged to be within the rounding applied, and therefore, not calculated."

The NRC staff did not review the SCAP-II code or methodology and therefore, its use is not found generically acceptable by this SE. However, based upon the WCNOC stated conformance with RG 1.183, Regulatory Position 4.2.1, the licensee's statements described above, and the NRC staff's experience, the staff finds that the estimated shine doses are reasonable and are therefore, acceptable for use in the determination of the WCNOC control room dose analyses.

3.5.5 Atmospheric Relative Concentration Estimates

3.5.5.1 Meteorological Data

Refer to Section 3.4.2 of this SE for discussion on the methodology used by the licensee to collect and format the meteorological data and results of the NRC staff review.

3.5.5.2 Control Room Atmospheric Dispersion Factors

Refer to Section 3.4.3 of this SE for discussion on the methodology used by the licensee to determine control room atmospheric dispersion factors and results of the NRC staff review.

3.5.5.3 EAB and LPZ Atmospheric Dispersion Factors

Refer to Section 3.4.4 of this SE for discussion on methodology used by the licensee to determine EAB and LPZ atmospheric dispersion factors and results of the NRC staff review.

3.5.6 Accident Dose Calculations

A full implementation of the AST is proposed for WCNOC. Therefore, to support the licensing and plant operation changes discussed in the LAR, the licensee analyzed the following accidents employing the AST as described in RG 1.183.

- Loss-of-Coolant Accident (LOCA)
- Fuel Handling Accident (FHA)
- Main Steam Line Break (MSLB) Accident
- Steam Generator Tube Rupture (SGTR) Accident
- Locked Rotor Accident (LRA) or Reactor Coolant Pump Shaft Seizure Accident
- Control Rod Ejection Accident (CREA)
- Letdown Line Break (LLB)
- Waste Gas Decay Tank (WGDT) Leak or Failure
- Radioactive Liquid Waste Tank (LWT) Leak or Failure
- Loss of Non-Emergency Alternating Current (AC) Power (LOAC)

3.5.6.1 Loss-of-Coolant Accident (LOCA)

The LOCA considered is a double-ended rupture of the largest pipe in the RCS. The objective of this postulated DBA is to evaluate the ability of the plant design to mitigate the release of radionuclides to the environment in the unlikely event that the ECCS is not effective in preventing core damage. A LOCA is a failure of the RCS that results in the loss of reactor

coolant that, if not mitigated, could result in fuel damage including a core melt. The primary coolant will blow down through the break into the containment, depressurizing the RCS and pressurizing the containment. A reactor trip occurs and the ECCS is actuated to force borated water into the reactor vessel. Containment sprays actuate to depressurize the containment. Thermodynamic analyses, using a spectrum of RCS break sizes, show that the ECCS and other plant safety features are effective in preventing significant fuel damage. Nonetheless, the radiological consequence portion of the LOCA analysis conservatively assumes that the ECCS is not effective and that substantial fuel damage occurs.

The licensee stated that the LOCA radiological analysis uses the analytical methods and assumptions outlined in RG 1.183, Appendix A, "Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident." A summary of the assumptions used in the analysis is provided in the column labeled "AST" of Table 4.3-12, "Assumptions Used for LOCA Analysis" in Enclosure IV (Reference 1), unless modified by supplemental information for this LAR such as the licensee's response to RAI ARCB1-CONTROL ROOM-3 (e.g. assumptions for calculating control room operator dose from transit to and from the control room building and the leakage to the RWST), RAI ARCB1-LOCA-1 (e.g. elimination of credit for sedimentation of particulate activity), and RAIs ARCB1-SGTR-6 and ARCB1-GENERAL-3 (e.g. atmospheric dispersion factors applied to control room unfiltered inleakage associated with ingress and egress while in the control room emergency mode). Also, the normal mode control building and control room inflow rates were increased by 10 percent in the supplemental responses to ARCB1-GENERAL-2 dated June 9 and December 6, 2018 and calculations were provided to determine the offsite doses without credit for the EES filters (see the response to ARCB1-LOCA-3 dated June 9, 2018).

3.5.6.1.1 Source Term for the LOCA

For the LOCA, the activity in the RCS is only modeled for the release from the mini-purge system. This is acceptable to the NRC staff because the RCS activity is negligible for the pathways where the source term includes the activity from melted fuel.

The gap release phase begins with the onset of fuel cladding failure at about 30 seconds and is assumed to continue for about 30 minutes. As the core continues to degrade, the gap release phase ends and the early in-vessel release phase begins. The early in-vessel phase continues for 1.3 hours. The inventory in each release phase is assumed to be released at a constant rate over the duration of the phase and starting at the onset of the phase. The LOCA source term release fraction, timing characteristics, and radionuclide grouping are tabulated in Table 3.5.6-1 of this SE.

Table 3.5.6-1
LOCA Release to Containment

Radionuclide Group	Gap Release Phase *(0.5 Hours)*[14]	Early In-Vessel Phase *(1.3 Hours)*
Noble Gases *(Xe, Kr, Rn, He)*	0.05	0.95
Halogens *(I, Br)*[15]	0.05	0.35
Alkaline Metals *(Cs, Rb)*	0.05	0.25
Tellurium Group *(Te, Sb, Se)*	0	0.05
Barium *(Ba, Sr)*	0	0.02
Noble Metals *(Ru, Rh, Pd, Mo, Tc, Co)*	0	0.0025
Cerium Group *(Ce, Pu, Np)*	0	0.0005
Lanthanides *(La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am)*	0	0.0002

The licensee assumes that the radionuclides released from the fuel are instantaneously and homogeneously distributed throughout the non-sprayed regions of the containment atmosphere as they are released from the fuel. The analysis credits mechanisms for removing released radionuclides from the containment atmosphere.

The licensee assumes that the radioiodine released to the containment atmosphere consists of 95 percent cesium iodide (CsI), 4.85 percent elemental radioiodine, and 0.15 percent organic forms. This radioiodine speciation is appropriate if the containment sump pH is maintained at a value of 7.0 or higher. This is accomplished at Wolf Creek by chemical injection.
Section 4.3.9.2.2.1, "Containment Leakage," of Enclosure IV states, in part that "[t]he sump pH is maintained at greater than or equal to 7.0," and, "[t]herefore, no re-evolution of iodine occurs." The licensee's analysis for the sump pH following a LOCA was evaluated by the NRC staff for acceptability in Section 3.3 of this SE.

[14] Bracketed values in this table provide duration of the release phase.
[15] Table 4.2-1 of Enclosure IV provides the core inventory used in the AST analyses including bromides, however bromides were not included in the dose calculations because WCNOC stated that they are expected to have no or little dose significance (see the licensee's supplemental response to ARCB1-CONTROL ROOM-5 provided in the letter dated January 15, 2018).

Furthermore, Section 4.4.2.6, "Results and Conclusions," of Enclosure IV, states, in part:

> In the calculation, the limiting system alignment of both Containment Spray trains operating with one NaOH eductor in service results in a sump pH of 7 in approximately 11 minutes, with the final long-term pH of 8.7 in approximately 70 minutes. The less limiting case of both Containment Spray trains operating with both NaOH supplies in service results in a sump pH of 7 in approximately 9 minutes, with the final long-term pH of 8.7 in approximately 35 minutes.

Section 4.4.2.6 of Enclosure IV also states the switchover to containment spray pump recirculation occurs at 32.73 minutes based upon the assumption of one educator operating.

3.5.6.1.1.1 Post-Accident Containment Sump Chemistry Management for the LOCA

In NUREG-1465 (Reference 42), the NRC staff concluded that iodine entering the containment from the RCS during an accident would be composed of at least 95 percent CsI, with no more than 5 percent of iodine (I) plus hydriodic acid. The radiation-induced conversion of iodide in water into elemental iodine is strongly dependent on the pH. The staff stated in the NUREG that without pH control, a large fraction of iodine dissolved in water pools in ionic form will be converted to elemental iodine and will be re-evolved into the containment atmosphere if the pH is less than 7. Conversely, if the pH is maintained above 7, less than 1 percent of the dissolved iodine will be converted to elemental iodine.

As discussed in Section 3.3 of this SE, the licensee conducted an evaluation of containment sump pH in order to ensure that particulate iodine deposited into the containment sump water does not re-evolve beyond the amount assumed in the DBA LOCA analysis. The NRC staff's evaluation of this analysis is provided in Section 3.5.6.1.2.

3.5.6.1.1.2 Summary of NRC Staff's Review of the Source Term for the LOCA

Based upon the discussion in Section 3.5.6.1.1 and the NRC staff's review of the licensee's analysis described in Section 3.5.6.1.1, the staff finds that the proposed source term assumptions are consistent with the guidance of RG 1.183 and are, therefore, acceptable.

3.5.6.1.2 Removal of Containment Atmosphere Radioactivity for the LOCA

The elemental iodine and particulate spray removal coefficients represent the rate at which elemental iodine vapor and particulate fission products are removed from the containment atmosphere by the spray system. In SRP, Section 6.5.2 (Reference 18.d), the effectiveness of the containment spray system can be estimated by considering the chemical and physical processes that can occur during an accident in which the system operates. Credit for the removal of elemental iodine and particulates starts at 2 minutes after the LOCA initiation. The licensee credits the maximum elemental iodine spray removal coefficient, λ_s, of 10 hour^{-1} consistent with the current licensing bases to conservatively address loading in the spray fluid during recirculation. The elemental iodine removal coefficient is credited until a decontamination factor (DF) of 200 is reached at approximately 2.33 hours. In addition, the licensee calculated a particulate spray removal coefficient, λ_p, of 5.0 hour^{-1} up to a DF of 50 (at 2.44 hours), and 0.5 hour^{-1} for DF's greater than 50 until the sprays are terminated at

9.5 hours.[16] The licensee also assumes no credit for the removal of particulates due to sedimentation.[17]

The NRC staff independently verified through calculations the licensee's particulate spray removal coefficients. Since these methods are consistent with SRP, Section 6.5.2, the staff finds that these values are acceptable.

3.5.6.1.3 Release Paths for the LOCA

Once dispersed in the containment, the release to the environment is assumed to occur through four pathways:

- Containment mini-purge at event initiation
- Release from containment leakage
- Sump water leakage from ECCS systems outside of the containment
- ECCS Back Leakage to RWST

3.5.6.1.3.1 Containment Mini-Purge at Event Initiation for the LOCA

The iodine activity in the RCS at the time of the accident is assumed to be at the TS limit of 1.0 microcurie/gm of DE I-131 for the maximum equilibrium RCS concentration. The noble gas activity concentration in the RCS at the time of the accident is assumed to be at the TS limit of 500 microcuries/gm of DE XE-133 for the maximum equilibrium RCS concentration. The activity for the remaining nuclide groups is at a 1 percent fuel defect level.

The licensee assumes that a containment purge is in progress at the start of the LOCA providing a path for releases to the environment. This purge is assumed to be isolated within 10 seconds as a result of a containment isolation signal. Since the onset of radionuclide releases from the fuel occurs at 30 seconds, this pathway is isolated prior to fuel damage occurring. The NRC staff's independent radiological analysis of this pathway assumed isolation of this purge pathway at 10 seconds. The minimum free volume of the containment modeled in this pathway is 2.5E6 cubic feet (ft^3). The only removal of activity from containment is by radioactive decay or the purge flow.

For purposes of analysis, the licensee assumes that the entire RCS inventory of volatile radionuclides is released to the containment and then it is released to the environment at a rate of 4680 cfm for 10 seconds. The licensee stated that the iodine release to the atmosphere is assumed to be 97 percent elemental and 3 percent organic iodine. The licensee stated that the assumed iodine chemical fractions do not impact the analyses results. Based upon the licensee's statement that these assumed forms of radioactivity do not impact the analyses results, the NRC staff finds this assumption acceptable.

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[16] Enclosure IV, Section 4.3.9.2.2.1, "Containment Leakage," credited spray until 5 hours following the accident initiation. The licensee revised this assumption in the June 19, 2018, supplemental response to RAI ARCB1-LOCA-1.

[17] Enclosure IV, Section 4.3.9.2.2.1, "Containment Leakage," credited sedimentation until a DF of 1000 was reached at 23.5. The licensee revised this assumption in the June 19, 2018, supplemental response to RAI ARCB1-LOCA-1.

3.5.6.1.3.2 Release from Containment Leakage for the LOCA

The maximum free volume of the containment modeled in the containment leakage pathway is 2.7E6 ft^3. The portion of this volume covered by spray drops (85 percent) and its unsprayed portion (15 percent) are modeled separately. The mixing rate between the sprayed and unsprayed regions is modeled as 69,400 cfm per fan cooler, which is more conservative than the current licensing basis value assumed. Only one fan cooler is assumed to be operating and a conservative delay of 2 minutes is assumed before mixing between the sprayed and unsprayed regions is credited. Mixing continues for the remainder of the event.

The LOCA radiological analyses assume that, within 60 seconds after the accident, isolation of the containment (except for the mini-purge isolation time, which is within 10 seconds), is complete and leakage terminated except for the design leakage rate.[18] Per the supplemental response to RAI ARCB1-LOCA-5 in the letter dated June 19, 2018 (Reference 10), the containment isolation total response time of 60 seconds includes the signal delay, diesel generator startup (for LOOP), and containment isolation valve stroke times. The containment is assumed to leak at the design leak rate of 0.2 percent per day for the first 24 hours of the accident and then to leak at half that rate (0.1 percent per day) for the remainder of the 30-day period considered in the analysis.

3.5.6.1.3.3 Sump Water Leakage from ECCS Systems Outside of the Containment for a LOCA

For the ECCS leakage pathway, all iodine activity released from the fuel is assumed to be in the sump solution immediately. The only removal of activity from the sump is by radioactive decay or leakage to the auxiliary building. The sump volume assumed is 460,000 gallons. When ECCS recirculation is established following a LOCA, leakage is assumed to occur from ESFs equipment in the auxiliary building. Recirculation is modeled to initiate at the start of the event and continues throughout the event.

The leakage to the auxiliary building is modeled at a rate of 2 gallons per minute (gpm). The leakage value was doubled in accordance with the guidance in RG 1.183, therefore, the proposed allowable design ESFs leakage is 1 gpm. The licensee confirmed that the calculated flashing fraction based on the maximum sump temperature, is less than 10 percent. Therefore, consistent with RG 1.183, the analysis assumes that 10 percent of the iodine activity in the leakage becomes airborne throughout the event and is available for release immediately to the environment. Recirculation is modeled to initiate at the start of the event and continues throughout the event. The licensee stated that the radioiodine that was postulated to be available for release to the environment was assumed to be 97 percent elemental and 3 percent organic and no reductions due to dilution or holdup were assumed.

The NRC staff held an audit on March 19 and 20, 2018 (audit summary documented by Reference 51). The audit, in part, discussed the calculations (including the LOCA) performed by the licensee to support the proposed changes to the TSs and the modifications to the design basis in accordance with 10 CFR 50.67. During the audit, the markup of TS LCO 3.7.13, "Emergency Exhaust System (EES)," provided in Attachment II of the supplemental information dated January 15, 2018, was discussed. The licensee provided this markup, in response to the NRC staff's RAI ARCB1-LOCA-3, to address an NRC staff concern regarding how the EES

[18] These isolation times are provided in the supplemental response to RAI ARCB1-LOCA-5 dated June 19, 2018.

boundary (e.g. auxiliary building boundary) is modeled in the proposed DBA consequences analyses.

In the revised TS LCO 3.7.13, the licensee proposed to add the words "that ensure the building boundary can be closed consistent with the safety analysis" to the NOTE associated with LCO 3.7.13; revise existing Required Action B.1, which states, "Restore auxiliary building boundary to OPERABLE status," with a 24-hour Completion Time, to Required Action B.1, which would state, "Initiate actions to implement mitigating actions," with an immediate Completion Time; add Required Action B.2, which would state. "Verify mitigating actions ensure main control room occupants do not exceed 10 CFR 50 Appendix A GDC 19 limits," with a 24-hour completion time; and add Required Action B.3, which would state, Restore building boundary to OPERABLE status," with a 24-hour Completion Time.

The NRC staff determines that the proposed markup to the NOTE associated with TS LCO 3.7.13 addresses the concern regarding the Note being consistent with the proposed radiological consequence analyses for control room and offsite doses. This will ensure that when the auxiliary building or fuel building boundary is opened intermittently under administrative controls, these administrative controls ensure that the building boundary can be closed consistent with the safety analysis. The NRC staff also agrees that the proposed markups to TS LCO 3.7.13, Required Action B.1 address the concern regarding the control room doses by verifying that mitigating actions ensure that the main control room occupants do not exceed 10 CFR Part 50 Appendix A, GDC 19 limits. The NRC staff's acceptance of the proposed changes to the Note and TS LCO 3.7.13, as discussed above, is based upon the reasonable assurance that the licensee's analysis assumptions for the LOCA analyses approved in this SE will ensure that the boundary can be closed consistent with the safety analysis (LCO 3.7.13 Note) or to determine if mitigating actions ensure the main control room occupants do not exceed 10 CFR Part 50 Appendix A, GDC 19 limits (LCO 3.7.13, Condition B Required Actions).

However, during the audit the NRC staff was concerned that the impact of the assumed modeling of the EES boundary on offsite doses had not been explicitly addressed by the proposed TS LCO markups. In order to address this concern, the NRC staff requested information regarding the TS LCO or the proposed calculation of offsite doses.

The licensee provided a supplemental response to RAI ARCB1-LOCA-3 dated June 19, 2018 (Reference 10). In the supplement the licensee stated that the only analysis that credits the EES for offsite doses is the LOCA dose calculation. The licensee also provided the results of a revised LOCA offsite radiological analysis where it did not credit the EES (consistent with TS LCO 3.7.13, Condition B). This analysis considered the loss of safety function for the EES in the proposed revised LOCA offsite radiological consequence analyses. Assuming the EES is not functioning (consistent with TS LCO 3.7.13, Condition B), the release point from the auxiliary building cannot be assured. Regarding the location of the release, the licensee stated that if the EES is not credited, the radioactivity will either leak out of the auxiliary building, remain within the building, or be exhausted from the unit vent. For the purpose of the offsite analysis, the release point would be from the auxiliary building or the unit vent. The licensee stated that with the EES not operating, as discussed above, the resulting doses remained within regulatory limits. Also, as stated, in part in Wolf Creek UFSAR Section 15.6.5.4.1.2, "Radioactive Releases Due to Leakage from ECCS and Containment Spray Recirculation Lines" (Reference 20):

No credit is taken for holdup or mixing in the auxiliary building; however, mixing and holdup in the containment sumps are included in the determination of radioactive material releases, and radioiodine removal through radioactive decay is credited for this leakage pathway.

The NRC staff has reviewed the supplemental responses to RAI ARCB1-LOCA-3 dated June 19, 2018 (Reference 10), and December 6, 2018 (Reference 14), and the licensee's proposed design basis assumption that the EES is not credited for the offsite dose analysis. In its letter dated December 6, 2018, the licensee described the assumptions used by the licensee to make a determination of compliance with 10 CFR 50.67. The response to RAI ARCB1-LOCA-3 also provided the licensee's results for this analysis for the EAB and LPZ doses. The NRC staff also performed an independent confirmatory calculation using the licensee's proposed assumptions including the assumption that the EES is not credited, as discussed above, for the offsite analysis.

The NRC staff's assessment confirmed that the licensee's conclusion with the EES not functioning, as described by TS 3.7.13 Condition B, that the 10 CFR 50.67 offsite limits would not be exceeded. The NRC staff's analysis and conclusion that the LOCA analysis results comply with applicable regulatory guidance discussed in Section 2.2 of this SE, are based in part on the licensee's:

(1) analysis assumption of a ground level release from the auxiliary building;

(2) analysis assumptions that no mixing or holdup is credited in the auxiliary building;

(3) analysis assumed release rates and atmospheric dispersion factors bounding any release rates from the auxiliary building;

(4) stated conformance with RG 1.194, Regulatory Position 3.2.4.2 (see the Table entitled, "Conformance with RG 1.194" of Enclosure IV);

(5) stated conformance with RG 1.183, Regulatory Position 5.1.3 (see Table A, "Conformance with RG 1.183" of Enclosure IV);

(6) offsite dose results provided in response to RAI ARCB1-LOCA-3 in the letter dated December 6, 2018, which do not exceed the accident dose values in 10 CFR 50.67;

(7) statements that the only analysis that credits the EES for offsite doses in the LOCA dose calculation; and

(8) the configuration and existence of any present auxiliary building structures, systems or components that have an impact on the assumptions made in the LOCA analysis (e.g. building walls or pathways to the environment.

3.5.6.1.3.4 ECCS Back Leakage to the RWST during a LOCA

For the RWST back-leakage pathway, a portion of the ECCS recirculation is assumed to leak into the RWST. All iodine activity released from the fuel is assumed to be in the sump solution immediately. The only removal of activity from the sump is by radioactive decay or leakage to the RWST. The sump volume assumed is 460,000 gallons. Recirculation is modeled to initiate at the start of the event and continues throughout the event.

The RWST is modeled at a rate of 2 gpm and is assumed to continue for 30 days.[19] The activity is modeled to be delivered directly to the gas filled portion of the RWST; however, only 10 percent of the activity becomes airborne and is available for release to the environment. The release rate from the RWST to the environment is based on the volume displacement from the incoming leakage. An adjustment is made to account for a reduction in the RWST gas volume available for dilution as the leakage into the RWST increases the water level.

3.5.6.1.4 Control Room Habitability for LOCA

At the start of a LOCA, the control room ventilation system is in normal mode. In this mode, unfiltered air from the environment enters the control building and control room. Receipt of a SI actuation signal or a high radiation signal from the control room air intake monitors will isolate the control room and initiate the emergency mode of operation, including a delay.

In the event of a LOCA, the low pressurizer pressure SI setpoint will be reached following the break. The SI signal or high radiation signal from the control room intake monitors will isolate the control room and cause the control room to switch from the normal operation mode to the emergency operation mode. The modeling of the switch to emergency mode is modeled at 120 seconds following event initiation, which includes a 60-second delay from the initiating signal.[20] As discussed in Sections 4.3.9.2.3, "Control Room" and 4.3.2.1, "Control Room Model," of Enclosure IV, operator action assumed to be taken 90 minutes after event initiation to isolate the failed train, reduces the unfiltered inflow to the control room, and consequently lowers the filtered inflow to the control building.

3.5.6.1.5 Conclusion for the LOCA

The licensee evaluated the radiological consequences resulting from the postulated LOCA and concluded that the radiological consequences at the EAB, LPZ, Control Room, and TSC are within the radiation dose reference values provided in 10 CFR 50.67 and the accident specific dose criteria specified in SRP, Section 15.0.1 (Reference 18.h). The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.2 of this SE. The licensee's calculated dose results are given in Table 3.5.7-1 of this SE. The NRC staff performed independent confirmatory dose evaluations, as necessary, to ensure a thorough understanding of the licensee's methods. The NRC staff concludes that the EAB, LPZ, Control Room, and TSC radiological doses, estimated by the licensee for the LOCA, meet the applicable accident dose criteria, and are therefore, acceptable.

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[19] Enclosure IV, Section 4.3.9.2.2.3, "Refueling Water Storage Tank Back-Leakage," assumed the leakage to the RWST to be 3.8 gpm. The licensee revised this leakage assumption to 2 gpm in response to RAI ARCB1-CONTROL ROOM-3 and RAI ARCB1-LOCA-4 by letter dated January 15, 2018. Also, in Table B of Enclosure IV, the licensee stated that the analysis conforms to RG 1.183, Appendix A, Regulatory Position 5.2, stating that the leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems that would require declaring the systems inoperable.

[20] Per WCNOC's response to RAI ARCB1-LOCA-2, dated January 15, 2018, the emergency mode, including filtration of the control room and control building inflow and recirculation flows, is modeled beginning at 120 seconds after the start of the LOCA.

3.5.6.2 Fuel Handling Accident (FHA)

A fuel assembly is assumed to be dropped and damaged during refueling, along with some of the fuel rods from a neighboring assembly. The FHA accident assumed to be most limiting is the dropping of a spent fuel assembly. The dropping of other loads, such as those allowed to be moved during core alterations (defined, in part, in TSs as the movement of fuel, sources or reactivity control components within the reactor vessel) were also considered as discussed in the June 19, 2018, supplemental response to RAI-ARCB1-FHA-3. Analysis of the accident is performed with assumptions selected so that the results are bounding for the accident occurring either inside containment or in the fuel building. The bounding activity pathway modeled releases of damaged fuel activity through the pool water to the building air space and then to the environment without crediting containment isolation or filtration by the fuel pool ventilation system. However, the EES (TS LCO 3.7.13), which serves both the auxiliary building and the fuel building is credited in the FHA analysis to ensure the release point assumed in the analysis is consistent with design and operation of the WCNOC facility.

The licensee stated that the FHA radiological analysis uses the analytical methods and assumptions outlined in RG 1.183, Appendix B, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident." A summary of the assumptions used are provided in the column labeled "AST" of Table 4.3-15, "Assumptions Used for Fuel Handling Accident Analysis" of Enclosure IV, unless modified by supplemental information for this LAR such as the licensee's response to RAI ARCB1-FHA-2 provided in the letter dated June 19, 2018 (e.g. the spent fuel pool overall decontamination). Also, the normal mode control building and control room inflow rates are increased by 10 percent, described in the supplemental response to RAIs ARCB1-GENERAL-2 and ARCB1-GENERAL-3, and the details of supplemental calculations used to determine the control room doses from releases from the containment personnel hatch through auxiliary building are provided in the supplemental responses to RAIs ARCB1-FHA-5, ARCB1-FHA-6 and ARCB1-GENERAL-3 by letter dated June 19, 2018.

3.5.6.2.1 Source Term for FHA

Appendix B of RG 1.183, Regulatory Position 1, identifies acceptable radiological source term assumptions for an FHA accident. Enclosure IV, Table C, "Conformance with Regulatory Guide 1.183, Appendix B (Fuel Handling Accident)," states that the FHA conforms to the RG 1.183, Appendix B, Regulatory Positions 1.1, 1.2 and 1.3 regarding the analysis source term used for the FHA.

The licensee assumed that all fuel rods in the equivalent of 1.2 fuel assemblies (one entire assembly and 20 percent of an adjacent assembly) are damaged to the extent that all their gap activity is released. The assembly inventory is based on the assumption that the subject fuel assembly has been operated at 1.65 times the core average power. The decay time used in determining the inventory of the damaged rods is 76 hours and is unchanged from the current licensing basis. The fission products release from the breached fuel is based on RG 1.183, Regulatory Position 3.2, "Release Fractions," and the estimate of the rods damaged. All the gap activity in the damage rods is assumed to be instantaneously released. Radionuclides that were considered include xenon, krypton, halogens, cesium and rubidium. The alkali metals (cesium and rubidium) were not modeled in the FHA dose calculation because they are assumed to be retained by the pool. Based upon the information provided by the licensee and the NRC staff's review of the source term assumptions, the NRC staff finds the FHA source term assumptions to be consistent with RG 1.183 and, therefore, acceptable.

The results of the NRC staff's review of the gap fractions assumed for the purposes of this analysis are provided in Section 3.6.7.1.2 of this SE.

3.5.6.2.2 Release Model for FHA

RG 1.183 allows credit for an overall pool DF for iodine of 200 for a pool depth of 23 feet. Although not explicitly discussed in RG 1.183, the specified overall DF of 200 applies to rod internal pressures up to 1,200 pounds per square inch gauge (psig). The assumption of 1,200 psig or less is not met for this amendment since the licensee's analysis assumes a maximum fuel rod internal pressure of 1,500 psig. The NRC staff previously acknowledged in RG 1.25, Revision 0, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," dated March 1972 (Reference 52)[21] that for release pressures greater than 1,200 psig, the DFs will be less than those in the regulatory guidance (currently is 200 in RG 1.183). RG 1.25 states that for pressures higher than 1,200 psig, the DF is to be calculated on an individual case basis using assumptions comparable to the conservatism used to derive the DF in the regulatory guidance. In the NRC staff's RAI ARCB-RAI-20, discussed in Enclosure VII and RAI ARCB1-FHA-2 in the letter dated December 4, 2017 (Reference 53), the NRC staff requested additional information to justify the use of a DF of 200 for fuel maximum internal pressures up to 1,500 psig. In response to RAI ARCB1-FHA-2 by letter dated June 19, 2018, the licensee justified a revised DF value of 170. The NRC staff reviewed the justification provided by the licensee, and independently assessed the DF of 170 for the licensee, for fuel with a maximum design pressure of 1,500 psig. The NRC's staff's approval of the DF of 170 is based, in part, upon the conservatism in the WCNOC FHA analysis and the assumption that the decay time of the fuel after shutdown (before a FHA could occur) is 76 hours.

The licensee assumed that 100 percent of the radionuclides released from the reactor cavity or spent fuel pool are released to the environment in 2 hours without any credit for filtration, holdup, or dilution. With the exception of different release points, the assumptions and inputs are identical for the FHA within the containment and the FHA outside the containment. To ensure that the analysis would be bounding for both release cases, the licensee did the analysis using the χ/Qs for most limiting combination of release point and receptor.

3.5.6.2.3 Single Failure

The current licensing basis of an FHA, for doses, accounts for a failure of the humidity controller in one train of the EES (described in Section 15.7.4.5.1.2(k) of the Wolf Creek UFSAR (Reference 20), which results in a decrease in filter efficiency to 82.5 percent. In the licensee's response to RAI ARCB1-FHA-7 by letter dated January 15, 2018, the licensee stated that in its revised analysis for offsite doses, no explicit single failure is modeled. In part, the licensee justified the proposed change in the single failure assumption in its response to RAI ARCB1-FHA-7, using the following statements and logic:

1. For the proposed FHA analysis, no credit is taken for EES filtration.

2. Due to the modeling for the FHA and the equipment required to be operable by TSs, there is not a single failure that will result in more limiting offsite doses.

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[21] RG 1.25 provides a historical perspective for the fuel rod pressures used to derive the DF of 200 in RG 1.183.

3. As stated in Wolf Creek TS LCO 3.7.13, two EES trains shall be operable during movement of irradiated fuel assemblies in the fuel building. Therefore, if a single failure of one train of EES is considered, the remaining second train of EES will be able to perform its associated safety function of transporting the radioactivity released within the fuel building to the unit vent, thereby maintaining the release point assumed for the dispersion factors used in the FHA. The licensee stated that regardless of if the humidity controller failure is considered or not, the EES filter efficiency credited in the analysis will be 0 percent and, therefore, the failure of a humidity controller would have no impact on the results of the analysis.

The NRC staff evaluated the proposed change in the single failure assumption for the FHA offsite dose analysis using RG 1.183, Regulatory Position 5.1.2, "Credit for Engineered Safeguard Features," which states, "[t]he single active failure that results in the most limiting radiological consequences should be assumed." Since the licensee stated that there is not a single failure that will result in a more limiting analysis, the staff evaluated whether the proposed method of evaluation would yield the limiting radiological consequences. The staff reviewed the logic and justification for the proposed change, including the licensee's response to RAI ARCB1-FHA-7. The NRC staff's acceptance of the proposed change to no longer assume a failure of the humidity controller in one train of EES is based on: (1) information provided by the licensee in its response to RAI ARCB1-FHA-7, (2) the assumption in the revised FHA analysis that takes no credit for EES, (3) the licensee's statement that there is not a single failure that will result in more limiting offsite doses, and (4) the existence of Wolf Creek TS LCO 3.7.13 requiring two trains of EES to be operable during movement of irradiated fuel assemblies in the fuel building.

3.5.6.2.4 Control Room Habitability for the FHA

Based on licensee's response to RAI ARCB1-GENERAL-3 (Table 3), by letter dated June 19, 2018, and December 6, 2018, the licensee assumes the control room ventilation system is actuated to the emergency mode of operation at 120 seconds from the event initiation. The control room unfiltered total inleakage during emergency mode is 50 cfm. Also, an operator action is assumed to be taken 90 minutes after the event initiation to isolate the ventilation train with the failed filtration.

3.5.6.2.5 TSC Habitability for the FHA

At the start of the FHA, the TSC system is in normal mode. During normal plant operation, unfiltered air from the environment enters the TSC at a flow rate of 550 cfm. Unfiltered inleakage during normal operation is 20 cfm.

The emergency mode of operation is assumed to be manually initiated within 60 minutes after the event initiation. During the emergency mode, the licensee assumed air from the environment enters the TSC at a flow rate of 550 cfm through filters. The assumed unfiltered inleakage is 20 cfm. The licensee also assumed 450 cfm of filtered recirculation flow. Air is discharged at a flow rate to match the total inflow to the compartment.

3.5.6.2.6 FHA in an Open Containment and Release to Control Room via the Auxiliary Building

The licensee provided a second FHA analysis, which considered the containment open and a release pathway from the containment to the control room via the auxiliary building. In response to RAIs ARCB1-FHA-5, ARCB1-FHA-6, and ARCB1-GENERAL-3, in its letter dated December 6, 2018, the licensee provided supplemental information describing the second FHA analysis and the assumptions that were changed from the FHA analysis discussed above in Section 3.5.6.2. These changes include, in part, crediting manual operator action at 30 minutes for the emergency mode actuation. The licensee's second FHA analysis results, described in the letter dated December 6, 2018, in response to RAI ARCB1-GENERAL-3, show that the control room doses are limiting (3.1 versus 1.1 rem TEDE) for the scenario, which considers a release to the control room via the auxiliary building. Using the licensee's assumptions, the NRC staff confirmed the licensee's results for the second FHA analysis.

3.5.6.2.7 Conclusion for the FHA

The licensee evaluated the radiological consequences resulting from the postulated FHA and concluded that the radiological consequences at the EAB, LPZ, control room, and TSC are within the radiation dose reference values provided in 10 CFR 50.67 and the accident specific dose criteria specified in SRP, Section 15.0.1. Unless, otherwise stated above, the NRC staff's review finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.2 of this SE. The licensee's calculated dose results are given in Table 3.5.7-1 of this SE. The NRC staff performed independent confirmatory dose evaluations, as necessary, to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, control room, and TSC radiological doses, estimated by the licensee for the FHA, meet the applicable accident dose criteria, and are therefore, acceptable.

3.5.6.3 Main Steam Line Break (MSLB)

The MSLB accident considered is the complete severance of an MSLB outside containment and is discussed in Wolf Creek UFSAR Section 15.1.5, "Steam System Piping Failure" (Reference 20). The affected SG will rapidly depressurize, and release radionuclides initially contained in the secondary coolant to the outside atmosphere. Primary coolant activity transferred to the affected SG via tube leakage will be released to the outside atmosphere as well. A portion of the activity initially contained in the intact SGs is released to the atmosphere through the ARVs and/or the MSSVs. A portion of the activity transferred to the intact SGs via tube leakage will also be released to the outside atmosphere. Using the proposed assumptions and models, a steam line break outside containment will bound a break inside containment since the outside break provides a means for direct release to the environment. The licensee stated that MSLB radiological analysis uses the analytical methods and assumptions outlined in RG 1.183, Appendix E, "Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break Accident." A summary of the assumptions used are provided in the column labeled "AST" of Table 4.3-6, "Assumptions Used for Main Steamline Break Analysis," provided in Enclosure IV, unless modified by subsequent supplemental information for this LAR such as the April 19, 2018, supplemental response to RAI ARCB1-MSLB-2.

Consistent with the current MSLB analysis, the proposed MSLB analysis assumes a loss of offsite power, a stuck rod cluster control assembly and the worst case single failure. The supplemental response to RAI ARCB1-MSLB-1, by letter dated June 19, 2018, confirms that the dose analysis models the effects of a LOOP concurrent with the MSLB and that this is conservative with respect to calculated radiological consequences.

The licensee stated that the analysis of the MSLB radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix E. The NRC staff's review is discussed below.

3.5.6.3.1 Source Term for MSLB

Appendix E of RG 1.183 identifies acceptable radiological analysis assumptions for an MSLB accident. RG 1.183, Appendix E, Regulatory Position 2, states, "[i]f no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications," and that "[t]wo cases of iodine spiking should be assumed." The licensee's evaluation indicates that no fuel damage would occur as a result of an MSLB accident. Therefore, consistent with RG 1.183, the licensee performed the MSLB accident analyses for two radioiodine spiking cases.

The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated MSLB that has raised the primary coolant iodine concentration to the maximum value permitted by the TS for a spiking condition. For Wolf Creek, the maximum iodine concentration allowed by TS LCO 3.4.16, as a result of an iodine spike, is 60 microcuries per gram of DE I-131.

The second case assumes that the primary system transient associated with the MSLB causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike. Initially, the plant is assumed to be operating with the RCS iodine activity at the TS limit for normal operation. For Wolf Creek, the RCS TS LCO 3.4.16 limit for normal operation is 1 microcurie per gram of DE I-131. The increase in primary coolant iodine concentration for the concurrent iodine spike case is estimated using a spiking model that assumes that as a result of the accident, iodine is released from the fuel rods to the primary coolant at a rate that is 500 times greater than the maximum iodine equilibrium release rate. The iodine release rate is also referred to as the iodine appearance rate. The concurrent iodine spike is assumed to persist for a period of 8 hours. The iodine appearance rates are calculated assuming maximum letdown flow with perfect cleanup. The licensee assumes that the activity released from the iodine spiking mixes instantaneously and homogeneously throughout the primary coolant system.

The alkali metal activity in the RCS at the time the MSLB occurs is at a 1 percent fuel defect level, and the activity in the secondary coolant is assumed to be in the same ratio as the primary-to-secondary iodine concentrations. Therefore, the secondary side alkali metal concentrations are assumed to be 10 percent of the primary alkali concentrations. The noble gas activity concentration in the RCS at the time the accident occurs is based on the TS value of 500 microcuries per gram of DE XE-133.

In accordance with RG 1.183, Appendix E, Regulatory Position 4, the licensee assumes the speciation for iodine release from the SGs is 97 percent elemental and 3 percent organic. In addition, the licensee included the radiological dose contribution from the release of secondary coolant iodine activity at the TS LCO 3.7.18 limit of 0.1 microcuries per gram of DE I-131, which

is 10 percent of the maximum equilibrium RCS concentration of 1.0 microcuries per gram of DE I-131.

Based upon the information provided by the licensee and the NRC staffs review of the source term assumptions, the staff finds the MSLB source term assumptions to be consistent with RG 1.183 and, therefore, acceptable.

3.5.6.3.2 Release Model for MSLB

The licensee followed the guidance as described in RG 1.183, Appendix E, Regulatory Position 5, in all aspects of the transport analysis for the MSLB.

The licensee stated that the accident-induced TS Bases leakage of 1 gpm was assumed to be entirely to the faulted SG and is immediately released to the atmosphere with no credit for iodine or alkali metal retention. The licensee also stated that the normal operation TS for primary-to-secondary leakage of 150 gallons per day (gpd)/SG was assumed to be evenly apportioned between the intact SGs.

RG 1.183, Appendix E, Regulatory Position 5.2, states:

> The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lb_m/hr [pound mass per hour]) should be consistent with the basis of the parameter being converted. The ARC [alternative repair criteria] leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc [gram per cubic centimeter] (62.4 lb_m/ft^3 [pound mass per cubic foot]).

The licensee stated that all primary-to-secondary leakage is modeled with density based on cooled liquid. The MSLB leak rate of 150 gpm per SG into the intact SGs before the accident, and the 1.0 gpm induced by the accident in the licensee's analysis is based upon cooled liquid and is assumed to be at a density of 62.4 lb_m/ft^3. Therefore, the NRC staff finds that the licensee complies with Regulatory Position 5.2.

RG 1.183, Appendix E, Regulatory Position 5.3, states:

> The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C [degrees Celsius] (212 °F) [degrees Fahrenheit]. The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

The licensee stated that 12 hours after the accident, the residual heat removal (RHR) system is assumed to be placed into service for heat removal and at this time there are no further steam releases to the atmosphere from the intact SGs. The licensee also states that within 34 hours after the accident, the RCS has been cooled to below 212 °F and there are no further steam releases to atmosphere from the faulted SG. Based upon these statements, the NRC staff agrees that the MSLB analysis conforms to Regulatory Position 5.3.

The licensee stated, and the NRC staff agrees that no reduction or mitigation of noble gas activity in the releases from the primary system was modeled. All noble gas activity carried over to the secondary side through the SG tube leakage is assumed to be immediately released to the outside atmosphere, and therefore, the analysis conforms to RG 1.183, Appendix E, Regulatory Position 5.4.

Following the guidance in RG 1.183, Appendix E, Regulatory Positions 5.5.1, 5.5.2, and 5.5.3, the licensee assumes that the faulted SG was assumed to blowdown to dry-out conditions and all of the primary-to-secondary leakage into the faulted SG flashes to vapor and is released to the environment with no mitigation. For the intact SGs that are used for plant cooldown, the licensee assumes that there is 150 gpd primary-to-secondary leakage to each SG in accordance with the normal limit. Leakage to the intact SGs was modeled to mix with the secondary water without flashing since the SG tubes are submerged.

RG 1.183, Appendix E, Regulatory Position 5.5.4, states:

> The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

Accordingly, the licensee assumes that the radioactivity in the initial bulk water of the unaffected SGs becomes vapor at a rate that is a function of the steaming rate and the inverse of the partition coefficient. The licensee used a partition coefficient of 100 for iodine released from the intact SGs.

The total release from the faulted SG is 165,000 lb_m initially (assumed to blow down over the first 2 minutes) plus 1.0 gpm from the primary-to-secondary leakage for 34 hours. Thirty-four hours after the accident, no further steam containing radionuclides is released from the faulted SG to the environment. The NRC staff finds the MSLB transport assumptions to be consistent with RG 1.183 and, therefore, acceptable.

3.5.6.3.3 Control Room Habitability for the MSLB

The assumptions utilized in modeling of the control room are discussed in Section 3.5.4, "Control Room and TSC" of this SE, unless modified by the licensee's supplemental response to RAI ARCB1-MSLB-2 by letter dated April 19, 2018. The licensee originally evaluated control room habitability for the MSLB assuming that the low steam line pressure safety injection setpoint would be reached almost immediately following the break and would facilitate the switch from the normal operation mode to the emergency operation mode. However, in response to the NRC staff's RAI ARCB1-MSLB-2, the licensee stated that below the P-11 interlock, manual SI actuation would be required to isolate the control room. Therefore, based upon a sensitivity analysis to conservatively estimate the control room doses, the time to isolate the control room was conservatively assumed to occur at 120 seconds following event initiation (concurrent with the assumed completion of the faulted SG blowdown). The license stated that relative to the completion of the faulted SG blowdown, earlier control room isolation reduces the activity ingress due to filtration, and later, control room isolation reduces the activity in the control room by exhausting it from the control room. Therefore, control room isolation coincident with the completion of the faulted SG blowdown was determined to be conservative. The normal intake damper and emergency mode actuation is assumed to occur at 120 seconds.

Operator action is assumed to be taken 90 minutes after event initiation to isolate the ventilation train with failed filtration.

3.5.6.3.4 TSC Habitability for the MSLB

At the start of the MSLB, the TSC system is in normal mode. During normal plant operation, unfiltered air from the environment enters the TSC at a flow rate of 550 cfm (500 cfm plus 10 percent uncertainty). Unfiltered inleakage during normal operation is 20 cfm.

The emergency mode of operation is assumed to be manually initiated within 60 minutes after the event initiation. During the emergency mode, the licensee assumed air from the environment enters the TSC at a flow rate of 550 cfm through filters. The assumed unfiltered inleakage is 20 cfm. The licensee also assumed 450 cfm (500 cfm minus 10 percent uncertainty) of filtered recirculation flow. Air is discharged at a flow rate to match the total inflow to the compartment.

3.5.6.3.5 Conclusion for the MSLB

The licensee evaluated the radiological consequences resulting from the postulated MSLB and concluded that the radiological consequences at the EAB, LPZ, control room, and TSC are within the radiation dose reference values provided in 10 CFR 50.67 and the accident specific dose criteria specified in SRP, Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.2 of this SE. The licensee's calculated dose results are given in Table 3.5.7-1 of this SE. The staff performed independent confirmatory dose evaluations, as necessary, to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, control room, and TSC radiological doses, estimated by the licensee for the MSLB, meet the applicable accident dose criteria, and are therefore, acceptable.

3.5.6.4 Steam Generator Tube Rupture (SGTR)

The SGTR accident considered is the complete severance of a single tube in one of the SGs resulting in the transfer of RCS water to the ruptured SG and is discussed in Wolf Creek's UFSAR Section 15.6.3, "Steam Generator Tube Rupture" (Reference 20). The primary-to-secondary break flow through the ruptured tube following the SGTR results in radioactive contamination of the secondary system. A portion of this radioactivity is released to the outside atmosphere through the main condenser, the ARVs, or the MSSVs. In addition, iodine and alkali metal activity is contained in the secondary coolant prior to the accident and is released to the atmosphere as a result of steaming from the SG following the accident. A summary of the assumptions used are provided in the column labeled "AST" of Table 4.3-11, "Assumptions Used for SGTR Dose Analysis" provided in Enclosure IV, unless modified by subsequent supplemental information such as the licensee's response to RAIs ARCB1-SGTR-2, ARCB1-SGTR-5, ARCB1-SGTR-6 and ARCB1-GENERAL-3 dated June 19, 2018 (Reference 10) and the licensee's response to RAI ARCB1-SGTR-1 dated December 6, 2018 (Reference 14).

In the supplemental response to RAI ARCB1-SGTR-2, dated June 19, 2018, the dose analysis models the effects of a LOOP concurrent with the SGTR. The scenario involves the release of steam from the secondary system in conjunction with loss of main steam dump capabilities and the subsequent venting to the atmosphere through the ARVs.

The limiting single failure relative to the radiological consequences is chosen in order to maximize the amount of radioactivity released to the atmosphere. The licensee stated that no single active failure scenario results in the flooding of the main steam lines. The licensee stated that the limiting single failure for the SGTR is the failure of the ARV to close on the loop with the ruptured SG. After identifying the stuck-open ARV, operations personnel are dispatched to locally close the ARV block valve. It is assumed that the block valve is closed within 30 minutes after the ARV becomes stuck open, thus terminating the release of radioactive steam from the ruptured SG to the atmosphere. Primary-to-secondary break flow will continue following the closure of the block valve until the primary and secondary system pressures are equalized. Equalization of the primary and secondary system pressures terminates the flow of reactor coolant into the ruptured SG in sufficient time to prevent filling it completely.

The time-dependent mass releases, used to assess the radiological consequences of this postulated SGTR, and provided in Table 4.3-11 of Enclosure IV, are superseded by those provided in the supplemental response dated June 19, 2018, to RAI ARCB1-SGTR-2. The primary-to-secondary leakage is assumed to be 1 gpm. Following the closure of the ruptured SG ARV block valve, there is an additional radiological dose due to the leakage from the primary system into the intact SGs, and the initial concentration of radioactivity contained in the intact SGs as steaming continues to provide plant cooldown.

The licensee states and the NRC staff agrees that the analysis of the SGTR radiological analysis uses the analytical methods and assumptions outlined in RG 1.183, Appendix F, "Assumptions for Evaluating the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident." The NRC staff's review is discussed below.

3.5.6.4.1 Source Term for SGTR

Appendix F of RG 1.183 identifies acceptable radiological analysis assumptions for an SGTR accident. RG 1.183, Appendix F, Regulatory Position 2, states that "[i]f no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specification," and that "[t]wo cases of iodine spiking should be assumed." The licensee's evaluation indicates that no fuel damage would occur as a result of a SGTR accident. Therefore, consistent with RG 1.183, the licensee performed the SGTR accident analyses for two radioiodine spiking cases.

The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated SGTR that has raised the primary coolant iodine concentration to the maximum value permitted by the TSs for a spiking condition. For Wolf Creek, the maximum iodine concentration allowed by TS LCO 3.4.16, as a result of an iodine spike, is 60 microcuries per gram of DE I-131.

The second case assumes that the primary system transient associated with the SGTR causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike. Initially, the plant is assumed to be operating with the RCS iodine activity at the TS limit for normal operation. For Wolf Creek, the RCS TS LCO 3.4.16 limit for normal operation is one microcurie per gram of DE I-131. The increase in primary coolant iodine concentration for the concurrent iodine spike case is estimated using a spiking model that assumes that as a result of the accident, iodine is released from the fuel rods to the primary coolant at a rate that is 335 times greater than the maximum iodine equilibrium release rate. The iodine release rate is also referred to as the iodine appearance rate. The concurrent iodine spike is assumed to persist for a period of 8 hours. The iodine appearance rates are calculated assuming maximum

letdown flow with perfect cleanup. The licensee assumes that the activity released from the iodine spiking mixes instantaneously and homogeneously throughout the primary coolant system.

The alkali metal activity in the RCS at the time the SGTR occurs is at a 1 percent fuel defect level and the activity in the secondary coolant is assumed to be in the same ratio as the primary-to-secondary iodine concentrations. Therefore, the secondary side alkali metal concentrations are assumed to be 10 percent of the alkali concentrations. The noble gas activity concentration in the RCS at the time the accident occurs is based on the TS value of 500 microcuries per gram of DE XE-133.

In accordance with RG 1.183, Appendix F, Regulatory Position 4, the licensee assumes the speciation for iodine release from the SGs is 97 percent elemental and 3 percent organic. In addition, the licensee included the radiological dose contribution from the release of secondary coolant iodine activity at the TS LCO 3.7.18 limit of 0.1 microcurie per gram of DE I-131, which is 10 percent of the maximum equilibrium RCS concentration of 1.0 microcurie per gram DE I-131.

Based upon the information provided by the licensee and the NRC staff's review of the source term assumptions, the staff finds the SGTR source term assumptions to be consistent with RG 1.183 and, therefore, acceptable.

3.5.6.4.2 Release Model for SGTR

The licensee followed the guidance as described in RG 1.183, Appendix F, Regulatory Position 5, in all aspects of the transport analysis for the SGTR. The NRC staff's review of the licensee's assumptions is discussed below.

RG 1.183, Appendix F, Regulatory Position 5.2, states:

> The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lb_m/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lb_m/ft^3).

In accordance with RG 1.183, Appendix F, Regulatory Positions 5.2, the licensee's SGTR primary-to-secondary leak rate of 1.0 gpm corresponds to a leakage density of 62.4 lb_m/ft^3. The primary-to-secondary leakage is assumed to be into the three intact SGs.

RG 1.183, Appendix F, Regulatory Position 5.3, states:

> The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

In accordance with RG 1.183, the licensee assumes that the primary-to-secondary break flow was terminated when the RCS pressure equalized with the ruptured SG secondary side pressure, and the release of radioactivity from the ruptured and intact SG was terminated at 12 hours when the RHR system was in service and removing all decay heat.

In the supplemental responses to RAI ARCB1-SGTR-2, dated June 19, 2018, and RAI ARCB1-SGTR-1, dated December 6, 2018, the dose analysis models the effects of a LOOP concurrent with the SGTR and the releases from the SGs to the atmosphere begin at the start of the SGTR event. The NRC staff finds that assuming the LOOP concurrent with the SGTR is consistent with RG 1.183, Appendix F, Regulatory Position 5.4 and is, therefore, acceptable to the NRC staff.

The licensee stated that no reduction or mitigation of noble gas activity in the releases from the primary system was modeled; all noble gas activity carried over to the secondary side through the SG tube leakage is assumed to be immediately released to the outside atmosphere as long as the steaming release continues, which according to Table 4.3-11 of Enclosure IV is 12 hours; and that the analysis conforms to RG 1.183, Appendix F, Regulatory Position 5.5.

The licensee stated that the SGTR analysis transport model conforms to Regulatory Positions 5.5 and 5.6 of Appendix E (see Regulatory Position 5.6 of Appendix F) for iodine and particulates. In addition, flashing of break flow in the ruptured SG with a time dependent flashing fraction was considered and all Iodine and alkali metal activity contained in the portion of the break flow that flashed to steam upon entering the ruptured SG is released directly to the atmosphere as long as steam releases from the ruptured SG continue. An iodine partition coefficient in the SGs of 100 (Curies iodine/gm water)/(Curies iodine/gm steam) is applied to releases resulting from steaming of the secondary side fluid.[22] The release of alkali metals from the secondary side is limited to the plant-specific moisture carryover factor of 0.25 percent.

The Wolf Creek specific transient SGTR analyses for thermal and hydraulic input, described in Section 2.7.3, "Steam Generator Tube Rupture – Input to Dose (UFSAR Section 15.6.3)," and the margin to overfill, described in Section 2.7.2, "Steam Generator Tube Rupture Margin to Overfill (UFSAR Section 15.6.3) in Enclosure I to letter dated January 17, 2017, confirm that the ruptured SG does not reach water levels that would result in tube uncover and that margin is maintained in the ruptured SG to prevent water relief via the MSSVs.

The major hazard associated with a SGTR event is the radiological consequences resulting from the transfer of radioactive primary coolant to the secondary side of the ruptured SG and subsequent release of radioactivity to the atmosphere. One major concern for an SGTR is the possibility of ruptured SG overfill because this could potentially result in a significant increase in the radiological consequences.

To address this concern, the licensee performed an analysis to demonstrate that the ruptured SG does not overfill and release water from the MSSVs. The analyses were performed to demonstrate that the secondary side of the ruptured SG did not completely fill with water. The SGTR margin to overfill transient analysis was performed using the RETRAN computer program and modifications were made to address Nuclear Safety Advisory Letter (NSAL)-07-11, "Decay

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[22] Per WCNOC's response to RAI ARCB1-SGTR-1, dated January 15, 2018, the partition coefficient of 100 is applied only to the radioactivity in the bulk water assumed to become vapor by steaming and not to any primary-to-secondary flow that flashed to steam.

Heat Assumption in Steam Generator Tube Rupture Margin-to-Overfill Analysis Methodology" (Reference 54), which identified a potential non-conservative assumption.

The accident modeled was a double-ended break of one SG tube located at the top of the tube sheet on the outlet (cold leg) side of the SG. The location of the break on the cold side of the SG results in higher primary to secondary break flow than a break on the hot side of the SG. It was also assumed that a LOOP occurs at the time of reactor trip, and the highest worth control rod assembly was assumed to be stuck in its fully withdrawn position at reactor trip.

The NRC staff reviewed the analysis results presented in Section 2.7.2 of Enclosure I of the LAR and concluded that overfill of the ruptured SG causing water to pass through the MSSVs will not occur for a design basis SGTR for WCGS.

3.5.6.4.3 Control Room Habitability for the SGTR

In the supplemental response to RAI ARCB1-SGTR-2, dated June 19, 2018, the licensee evaluated the control room habitability for the SGTR assuming that the control room ventilation system automatically transfers to the emergency filtration mode 60 seconds into the event. Also, in the response to RAI ARCB1-SGTR-2 the licensee stated that the 60 second value bounds process rack time (< 0.3 seconds), the solid-state protections signal time (< 2 seconds), the master/slave relay and diesel start time (< 12.2 seconds), and the damper closure time (< 30 seconds) and approximately the remaining 15 seconds is added for conservatism.

In the RAI ARCB1-GENERAL-3 responses, by letters dated June 19, 2018, and December 6, 2018, the licensee assumed that the total unfiltered inleakage modeled during emergency mode is 50 cfm. Prior to the closure of the normal HVAC intake damper, the 50-cfm unfiltered inleakage is associated with the normal HVAC intake atmospheric dispersion factor. After closure of the normal HVAC intake damper, the unfiltered inleakage is apportioned between the emergency mode HVAC intake (40 cfm) and the communications corridor intake (10 cfm associated with ingress/egress). Also, the licensee assumed that the time to generate the SI signal will be modeled to occur at 325 seconds after the event initiation and the normal intake damper closes at 10 minutes from event initiation. Also, operator action is assumed to be taken 90 minutes after event initiation to isolate the ventilation train with failed filtration. Other assumptions utilized in modeling of the control room are discussed in Section 3.5.4 of this SE unless modified by the licensee's supplemental responses.

3.5.6.4.4 TSC Habitability for SGTR

At the start of the SGTR, the TSC system is in normal mode. During normal plant operation, unfiltered air from the environment enters the TSC at a flow rate of 550 cfm. Unfiltered inleakage during normal operation is 20 cfm.

The emergency mode of operation is assumed to be manually initiated within 60 minutes after the event initiation. During the emergency mode, the licensee assumed air from the environment enters the TSC at a flow rate of 550 cfm through filters. The assumed unfiltered inleakage is 20 cfm. The licensee also assumed 450 cfm of filtered recirculation flow. Air is discharged at a flow rate to match the total inflow to the compartment. Additional inputs for the AST TSC model are provided in Table 4.3-16 of Enclosure IV.

3.5.6.4.5 Conclusion for the SGTR

The licensee evaluated the radiological consequences resulting from the postulated SGTR and concluded that the radiological consequences at the EAB, LPZ, control room, and TSC are within the radiation dose reference values provided in 10 CFR 50.67 and the accident specific dose criteria specified in SRP, Section 15.0.1. The NRC staff's review finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.2 of this SE. The licensee's calculated dose results are given in Table 3.5.7-1 of this SE. The staff performed independent confirmatory dose evaluations, as necessary, to ensure a thorough understanding of the licensee's methods. The staff finds that the EAB, LPZ, control room, and TSC radiological doses, estimated by the licensee for the SGTR, meet the applicable accident dose criteria, and are therefore, acceptable.

3.5.6.5 Locked Rotor Accident (LRA)

The LRA accident is described in the UFSAR Section 15.3.3, "Reactor Coolant Pump Shaft Seizure (Locked Rotor)" (Reference 20). For this DBA, an RCP rotor is assumed to seize instantaneously causing a rapid reduction in the flow through the affected RCS loop. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low flow signal. The licensee's evaluation indicates that fuel cladding damage will occur as a result of this accident. The flow imbalance creates localized temperature and pressure changes in the core. With the coincident LOOP, the condensers are not available, so the excess heat is removed from the secondary system by a steam dump through the steam generator safety valves and ARVs.

The licensee has analyzed the LRA radiological consequences by applying the release pathway models and assumptions outlined in RG 1.183, Appendix G, "Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident," and Appendix E, "Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break." Therefore, the NRC staff finds the licensee's analysis acceptable.

3.5.6.5.1 Source Term for the LRA

The licensee's LRA analysis postulates fuel damage. The limiting LRA radiological consequence analysis assumes 5 percent of the fuel rods become damaged by the event. The core source term described in Section 3.5.1.1, "Fission Product Inventory," of this SE is applicable to this event. The fraction of these activities available for release into the coolant are based upon the gap inventory fractions shown in RG 1.183, Table 3, "Non-LOCA Fraction of Fission Product Inventory in Gap," for non-LOCA gap inventory fractions with an assembly radial peaking factor of 1.65. Radionuclides released from the fuel are instantaneously and homogeneously distributed throughout the primary coolant. The licensee assumes that the chemical form of iodine released from the SGs to the environment is 97 percent elemental and 3 percent organic and is consistent with RG 1.183, Appendix E, Regulatory Position 4.

3.5.6.5.2 *Release Transport for the LRA*

The licensee followed the guidance as described in RG 1.183, Appendix G, Regulatory Position 5, in all aspects of the transport analysis for the LRA analysis. The radiological consequences are due to leakage of the contaminated reactor coolant to the SGs and from there, the environment. The primary-to-secondary leak rate for all four SGs is 1 gpm. The NRC staff finds that the licensee's assumption regarding the primary-to-secondary leak rate is

consistent with the leakage performance criteria of the SG program described in the TSs and, therefore, is acceptable. A LOOP is conservatively assumed to occur at the start of the LRA, rendering the main condenser unavailable for steam dump. With the main condenser unavailable, the plant is cooled down by releases of steam to the environment.

In accordance with the guidance in RG 1.183, the licensee assumes that all noble gas radionuclides released from the primary system are released through the SG to the environment without reduction or mitigation. Following the guidance in RG 1.183, Appendix G, Regulatory Position 5.6, which directs the use of the assumptions in Regulatory Positions 5.5 and 5.6 of RG 1.183 Appendix E, the licensee utilized RG 1.183, Appendix E, Regulatory Positions 5.5 and 5.6 to model the transport of radioiodine and particulate releases from the SGs. The licensee assumes that the radioactivity in the initial bulk water of the SG becomes vapor at a rate that is a function of the steaming rate and the inverse of the partition coefficient. The licensee used a partition coefficient of 100 for iodine released from the SG. The plant-specific moisture carryover factor of 0.25 percent to the steam releases is applied to the release of alkali metals from the secondary side. The release to the environment continues until 12 hours when the RCS temperature is cooled to 212 °F. The NRC staff finds that the licensee's assumptions regarding the transport and duration of the primary-to-secondary leakage to be consistent with Appendix E of RG 1.183 and therefore acceptable.

3.5.6.5.3 Control Room Habitability for the LRA

At the start of the event, the control room ventilation system is in normal mode. In this mode, unfiltered air from the environment enters the control building and control room. Receipt of a SI actuation signal or a high radiation signal from the control room, air intake monitors will isolate the control room and initiate the emergency mode of operation, including a delay. The analysis assumes no delay in the release and transport of radioactivity through the primary and secondary systems or in the transport from the release point to the air intake. The estimated radioactivity concentration at the detector immediately following event initiation is 7.4E-3 curies per cubic meter. The analysis models a detector setpoint of 2.12E-3 curies per cubic meter. Therefore, an instantaneous generation of the high radiation signal is assumed. The control room ventilation system enters emergency mode operation 120 seconds into the event, which includes a 60-second delay from the control room air intake monitor initiating signal.

After an emergency mode is initiated, outside air is brought into the control building through safety grade filters. Makeup air is brought into the control room via both trains of the control room filtration system that draws in air from the control building. Unfiltered air leaks into the control room and control building for the duration of the event. In addition, a filtered recirculation flow is modeled for the control room during emergency mode operation. Air in the control room and control building is discharged at flow rates to match the total inflow to the compartments. Operator action is assumed to be taken 90 minutes after event initiation to isolate the ventilation train with failed filtration.

3.5.6.5.4 TSC Habitability for the LRA

At the start of the SGTR, the TSC system is in normal mode. During normal plant operation, unfiltered air from the environment enters the TSC at a flow rate of 550 cfm. Unfiltered inleakage during normal operation is 20 cfm.

The emergency mode of operation is assumed to be manually initiated within 60 minutes after the event initiation. During the emergency mode, the licensee assumed air from the

environment enters the TSC at a flow rate of 550 cfm through filters. The assumed unfiltered inleakage is 20 cfm. The licensee also assumed 450 cfm of filtered recirculation flow. Air is discharged at a flow rate to match the total inflow to the compartment.

3.5.6.5.5 Conclusion for the LRA

The licensee evaluated the radiological consequences resulting from the postulated locked rotor and concluded that the radiological consequences at the EAB, LPZ, control room, and TSC are within the radiation dose reference values provided in 10 CFR 50.67 and the accident specific dose criteria specified in SRP, Section 15.0.1. The NRC staff's review finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.2 of this SE. The licensee's calculated dose results are given in Table 3.5.7-1 of this SE. The NRC staff performed independent confirmatory dose evaluations, as necessary, to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, control room, and TSC radiological doses, estimated by the licensee for the locked rotor, meet the applicable accident dose criteria, and are therefore, acceptable.

3.5.6.6 Control Rod Ejection Accident (CREA)

The CREA accident considered is a mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a RCCA and drive shaft and is discussed in the Wolf Creek UFSAR Section 15.4.8.3, "Radiological Consequences" (Reference 20). As a result of the accident, some fuel cladding damage and a small amount of fuel melt are assumed to occur. Due to the pressure differential between the primary and secondary systems and the assumed SG tube leakage, radioactive reactor coolant passes from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through the ARVs and/or the MSSVs. Finally, radioactive reactor coolant is discharged to the containment via the opening in the reactor vessel head. A portion of this radioactivity is released through containment leakage to the environment.

The licensee stated that the analysis of the CREA radiological analysis uses the analytical methods and assumptions outlined in RG 1.183, Appendix H, "Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident." A summary of the assumptions used are provided in the column labeled "AST" in Table 4.3-9, "Assumptions Used for Rod Ejection Analysis," provided in Enclosure IV, unless modified by subsequent supplemental information for this LAR such as the licensee's response to RAI ARCB1-SGTR-6 and ARCB1-GENERAL-3 (Table 2, "Communications Corridor Atmospheric Dispersion Factors χ/Q") dated June 19, 2018, which provided the atmospheric dispersion factors used for releases from the unit vent (denoted as MSSVs) to the control room ingress/egress pathway during the control room emergency mode. The supplemental information in the licensee's response to ARCB1-GENERAL-2, dated June 19, 2018, also increases the normal mode control building and control room inflow rates by 10 percent.

Following the applicable guidance, the licensee evaluated two separate release scenarios for the CREA involving a release from the containment and a release from the secondary system as discussed below. Both CREA release scenarios are analyzed with the assumption of a concurrent LOOP[23] and the worst case single failure.

[23] See the supplemental response to RAI ARCB1-CREA-1 dated June 19, 2018.

3.5.6.6.1 Source Term for the CREA

The source term for the CREA assumes fuel damage consisting of localized damage to fuel cladding and fuel melt occurring in the damaged rods. Source term release fractions for the CREA are described in RG 1.183, Appendix H, Regulatory Position 1, which states,

> Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.

The licensee assumed that as a result of the CREA, 10 percent of the fuel rods in the reactor core fail or experience departure from nucleate boiling resulting in cladding damage and releasing all of their gap activity. A small fraction of the fuel in the damaged fuel rods is assumed to melt as a result of the CREA equal to 0.25 percent of the core. A peaking factor of 1.65 was applied to the fission product inventory of the damaged rods. Consistent with the guidance provided in RG 1.183, Appendix H, Regulatory Position 1, for the breached fuel, the licensee assumed that 10 percent of the core inventory of noble gases and iodine reside in the fuel gap and will be available for release in both the containment release scenario and the secondary-side release scenario. The analysis also modeled fuel melting release fractions of 100 percent of noble gases, and 50 percent of iodine and alkali metals in the melted fuel rods for both release scenarios. Note that this release fraction assumption is conservative compared to the Appendix H, Regulatory Position 1 for the containment release pathway.

In accordance with RG 1.183, Appendix H, Regulatory Position 3, 100 percent of the released activity is assumed to be released instantaneously and mixed homogeneously throughout the containment atmosphere for the containment release scenario; and, 100 percent of the released activity is assumed to be released instantaneously and completely dissolved in the primary coolant and available for release to the secondary containment in the secondary-side release scenario. Consistent with RG 1.183, Appendix H, Regulatory Position 4, the iodine chemical fractions for releases to containment were modeled as 95 percent CsI, 4.85 percent elemental, and 0.15 percent organic. All fission products, with the exception of elemental and organic iodine and noble gases, were assumed to be in particulate form.

Appendix H, Regulatory Position 4, states, in part that "if the containment sump pH is not controlled at values of 7 or greater, the iodine species should be evaluated on an individual case bases." The licensee stated that no removal processes were modeled in containment besides leakage and decay, so the sump pH has no impact. The NRC staff finds the assumed chemical forms of iodine released to the containment atmosphere acceptable based upon the licensee's statements that no removal processes were modeled in containment and the licensee's assumptions for the CREA that all the filter efficiencies for all forms of iodine and particulates are the same (95 percent) for the: (1) filtered flow into the control building, (2) filtered flow for the control room (for the makeup to the control room from the control building and for the control room filtered recirculation flow), and (3) filtered flow into the TSC (for the makeup to the TSC and for the TSC recirculation flow). The basis of the staff's acceptance of

the licensee's assumed iodine chemical fractions is that, using the licensee's assumptions, the calculated doses would be independent of the assumed chemical fractions released to the containment.

The licensee assumes that the chemical form of iodine released from the SGs to the environment is 97 percent elemental and 3 percent organic and this assumption is consistent with RG 1.183, Appendix H, Regulatory Position 5.

The NRC staff finds the licensee's CREA source term assumptions to be consistent with (or more conservative than) the RG 1.183, Appendix H Regulatory Positions on the CREA source term and, therefore, are acceptable.

3.5.6.6.2 Transport from Containment for the CREA

The licensee assumes that the activity released to the containment mixes instantaneously throughout the free volume of containment with no credit assumed for removal of iodine or noble gas in the containment due to containment sprays or natural deposition. The licensee assumes that all containment leakage to the environment is assumed to be at the TS limit of 0.2 percent of containment air weight per day for the first 24 hours and 0.1 percent per day thereafter. The NRC staff finds the CREA containment transport assumptions to be consistent with RG 1.183 and, therefore, acceptable.

3.5.6.6.3 Transport from Secondary System for the CREA

In accordance with RG 1.183, Appendix H, Regulatory Position 7, the licensee evaluated the transport of activity from the RCS to the SGs secondary side assuming the total accident induced primary-to-secondary leak rate of 1.0 gpm is apportioned equally to all four SGs. The licensee assumes that the primary-to-secondary leakage was terminated when the release of radioactivity from the SGs was terminated. This occurred at 12 hours when the RHR system is assumed to be placed into service for heat removal. In accordance with RG 1.183, Appendix H, Regulatory Positions 7.2 and 7.3, the licensee assumes that the density used in converting volumetric leak rates to mass leak rates is 1.0 gram per cubic centimeter and that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation.

The licensee stated that the CREA analysis transport model conforms, as appropriate, to RG 1.183, Regulatory Positions 5.5 and 5.6 of Appendix E (see Regulatory Position 7.4 of Appendix H) for iodine and particulates. In Enclosure I to the licensee's letter dated January 15, 2018, the licensee's response to RAI ARCB1-CREA-2 further clarified how the Appendix E Regulatory Positions 5.5 and 5.6 were considered in the Wolf Creek CREA analyses. Following the guidance from RG 1.183, Appendix E, Regulatory Position 5.5, the licensee assumes that all of the primary-to-secondary leakage in the SGs mix with the secondary water without flashing. For iodine, because the SG tubes remain covered for the duration of the CREA, the partition coefficient of 100 was taken directly from RG 1.183. The licensee's assumed LOOP coincident with the start of the CREA makes the condenser unavailable and the reactor cooldown is achieved using steam releases from the SGs until initiation of shutdown cooling.

The licensee also proposed crediting a revised total SG mass after 2 hours (from the current value of 4.16E+5 lb_m to 4.85E+5 lb_m). In the licensee's response to the NRC staff's

RAI APHB-1 in the letter dated January 15, 2018, the licensee stated that the SG water level would automatically reach this level without operator action within a couple of minutes of reaching the SG low setpoint. The licensee also stated that there is not a single failure for the CREA that will result in a loss of safety function of auxiliary feedwater and that the SG water level will remain above 6 percent of the SG unless multiple failures are considered (i.e., a beyond design basis event, which is not considered in the design basis CREA). Therefore, the NRC staff finds this change to the total SG mass after 2 hours acceptable based upon the RAI response and because the new mass would be achieved automatically with no operator action necessary to maintain the SG level associated with the revised mass.

Based upon the NRC staff's review, described in part above, the NRC staff finds the CREA secondary system transport assumptions to be consistent with the guidance in RG 1.183 and, therefore, acceptable.

3.5.6.6.4 Control Room Habitability for the CREA – Releases to Containment Atmosphere

The licensee evaluated control room habitability for the CREA assuming that the low steam line pressure SI setpoint will be reached within 150 seconds of the event initiation due to the loss of RCS mass through the hole in the upper head. The NRC staff finds the assumed 150 second time to be acceptable based upon the statement in the UFSAR Section 15.4.8.2.2, "Calculation of Basic Parameters" (Reference 20), for the "Spectrum of Rod Cluster Control Assembly Ejection Accidents," that states: "The safety injection system is actuated on low pressurizer pressure within 1 minute after the break," based upon depressurization calculations for a typical four-loop plant and assuming the maximum possible size break.

The SI signal causes the control room to switch from the normal operation mode to the emergency operation mode. The licensee stated that the switchover to the emergency mode of operation is modeled at 210 seconds following event initiation, which includes a 60-second delay from the initiating signal. Also, operator action is assumed to be taken 90 minutes after event initiation to isolate the ventilation train with failed filtration.

3.5.6.6.5 Control Room Habitability for the CREA – Releases to the Secondary

Per the licensee, for the primary-to-secondary leakage case, there is no SI signal generated. However, the licensee determined that the control room intake radiation monitor signal setpoint is reached instantaneously based upon the model, which assumes no delay in the release of activity and no delays in the transport of activity through the primary and secondary systems or in the transport from the release point to the air intake. As a result, the control room ventilation system is assumed to enter the emergency mode operation 120 seconds into the event, which includes a 60-second delay from the control room air intake monitor initiating signal and 60 seconds for the switchover to occur. Because the accident relies solely on the control room air intake monitor for control room isolation, the control room unfiltered inleakage is assumed to continue to enter through the normal mode receptor for the duration of the accident. This is conservatively modeled to continue for the duration of the accident. Also, operator action is assumed to be taken 90 minutes after event initiation to isolate the ventilation train with failed filtration.

3.5.6.6.6 TSC Habitability for the CREA

At the start of the CREA, the TSC system is in normal mode. During normal plant operation, unfiltered air from the environment enters the TSC at a flow rate of 550 cfm. Unfiltered inleakage during normal operation is 20 cfm.

The emergency mode of operation is assumed to be manually initiated within 60 minutes after the event initiation. During the emergency mode, the licensee assumed air from the environment enters the TSC at a flow rate of 550 cfm through filters. The assumed unfiltered inleakage is 20 cfm. The licensee also assumed 450 cfm of filtered recirculation flow. Air is discharged at a flow rate to match the total inflow to the compartment.

3.5.6.6.7 Conclusion for the CREA

The licensee evaluated the radiological consequences resulting from the postulated CREA and concluded that the radiological consequences at the EAB, LPZ, control room, and TSC are within the radiation dose reference values provided in 10 CFR 50.67 and the accident specific dose criteria specified in SRP, Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.2 of this SE. The licensee's calculated dose results are given in Table 3.5.7-1 of this SE. The NRC staff performed independent confirmatory dose evaluations, as necessary, to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, control room, and TSC radiological doses, estimated by the licensee for the CREA, meet the applicable accident dose criteria, and are therefore, acceptable.

3.5.6.7 Letdown Line Break (LLB)

The analysis models the complete severance of the letdown line outside of containment. The occurrence of a complete severance of the letdown line would result in a loss of reactor coolant outside the primary containment. The LLB analysis is a bounding analysis that represents all small lines transporting reactor coolant inventory outside of containment as it poses the most severe consequences regarding radioactivity release based upon break size. The LLB is not discussed in an Appendix to RG 1.183; however, the licensee applied the AST discussed in RG 1.183 to this analysis in conjunction with event guidance from SRP, Section 15.6.2.

3.5.6.7.1 Source Term for the LLB

The licensee determined that no fuel failure results from the LLB. Consistent with SRP, Section 15.6.2, the analysis was performed with the RCS iodine activity at the limit allowed by the TS, including the effects of a concurrent iodine spike. The iodine spike increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the appearance rate corresponding to the maximum equilibrium RCS concentration of 1.0 microcurie per gram of DE I-131. The duration of the iodine spike is 8 hours.

However, Wolf Creek TS LCO 3.4.16, "RCS Specific Activity," requires the "RCS DE I-131 and DE XE-133 specific activity shall be within limits." Specifically, iodine must be less than or equal to 1.0 microcurie per gram and, if exceeded, TS LCO 3.4.16, Condition A, allows 48 hours to restore iodine to below the limit as long as DE I-131 is verified to remain equal to or below 60 microcuries per gram. TS LCO 3.4.16 assumes the initial reactor coolant iodine activity at 60 microcuries per gram of DE I-131 due to a pre-accident iodine spike caused by an RCS transient. It is possible for a reactor transient to occur prior to the postulated LLB; therefore, the

radiological consequences from this iodine spike should be evaluated for the LLB. In the letter dated December 4, 2017 (Reference 53), the NRC staff requested that the licensee submit for the NRC staff's review an analysis or a description of the LLB radiological consequences analysis assuming a pre-accident iodine spike. In the letter dated January 15, 2018, the licensee stated:

> A pre-accident iodine spike is not required to be analyzed for the Letdown Line Break event. Radiological consequences for the failure of small lines carrying primary coolant outside containment (i.e. Letdown Line Break) are not described in Regulatory Guide 1.183, but guidance is provided by NUREG-0800, the Standard Review Plan (SRP) Sections 15.0.3 and 15.6.2. SRP Section 15.0.3 provides guidance on design basis accident radiological consequence analyses for advanced light water reactors, and refers to SRP Section 15.6.2 for guidance on the Small Line Break event. SRP Section 15.6.2 describes an accident-initiated (also called coincident) iodine spike, and does not describe a pre-accident iodine spike. Additionally, SRP Section 15.0.3, Table 1 defines a limit of 2.5 rem TEDE for the Small Line Break; this limit is consistent with other event scenarios that consider an accident-initiated iodine spike, such as Main Steamline Break and Steam Generator Tube Rupture. The pre-accident spike scenarios for those events have a higher limit of 25 rem TEDE. Because the regulatory guidance is silent on a pre-accident spike for a small line break, and the SRP provides guidance for the Small Line Break that does not include a pre-accident iodine spike, a pre-accident iodine spike was not analyzed for the LLB event.

During the regulatory audit on March 19 and 20, 2018, the NRC staff explained to the licensee that the Wolf Creek TSs allow operation with a pre-accident iodine spike level and that a technical basis is needed for the current TS or the TS should be revised consistent with the new proposed LLB analysis. The licensee acknowledged the TSs allowed operation with a pre-accident spike level and by letter dated June 19, 2018, the licensee provided a revised LLB analysis, which included a pre-accident iodine spike. A pre-accident iodine spike was assumed to occur prior to the postulated LLB that has raised the primary coolant iodine concentration to the maximum value permitted by the TS for a spiking condition. For Wolf Creek, the maximum iodine concentration allowed by TS LCO 3.4.16, as the result of an iodine spike, is 60 microcuries per gram of DE I-131.

The noble gas activity concentration in the RCS at the time the accident occurs is based on the TS E3.4.16 value of 500 microcuries per gram of DE XE-133. The alkali metal activity concentration in the RCS is at a 1 percent fuel defect level. Iodine available for release to the atmosphere is assumed to be 97 percent elemental and 3 percent organic.

3.5.6.7.2 Transport for the LLB

Upon occurrence of the LLB, the reactor coolant is assumed to be released at a rate of 141 gpm until the isolation valve is fully closed. In a letter dated December 4, 2017 (Reference 53), the NRC staff pointed out that the current LLB analysis assumes that the loss of reactor coolant is 444 gpm and asked the licensee to justify the new assumed break flow of 141 gpm. In its letter dated January 15, 2018, the licensee stated:

> The 141 gpm value corresponds to the maximum break flow corresponding to the limiting letdown lineup procedurally allowed during normal operation....

> During normal operation, the letdown flow is 75 or 120 gpm, and one mixed bed demineralizer is in service... During normal operation with maximum purification, the letdown flow is 120 gpm...

> As such, a letdown line break flow of 141 gpm is expected to bound all letdown line breaks that could occur during normal operation.

> The current licensing basis [CLB] analysis is based upon a letdown line break flow of 222 gpm. The 444 gpm value is based upon doubling the break flow of 222 gpm ... The 222 gpm break flow corresponds to a letdown lineup that is not procedurally allowed during normal operation, i.e., it assumes all letdown lines are in use prior to the break occurring.

> ...Nevertheless, in order to ensure a bounding Letdown Line Break analysis, the 141 gpm value used for the Alternative Source Term (AST) Letdown Line Break analysis will be increased to 222 gpm to be consistent with the CLB.

During the regulatory audit conducted on March 19 and 20, 2018, the NRC staff expressed a concern associated with the LLB flow rate assumed in the dose analysis. Specifically, the concern was associated with the fact that the letdown flow rate of 222 gpm was not doubled as currently described in Section 15.6.2 of the Wolf Creek UFSAR to account for the backflow at the break site. The licensee discussed that the letdown heat exchanger would reduce the temperature of water downstream of the break below 212 °F and prevent the water that backflows from flashing. However, the NRC staff pointed out that while the water would be below 212 °F, RG 1.183, Appendix A, Regulatory Position 5.5, states that if the water is below 212 °F, then the flashing fraction should be assumed to be 10 percent, unless a lower value can be justified. Thus, NRC staff requested that either additional information be provided to justify not doubling the assumed break flow rate or revise the LLB analysis with a doubled assumed break flow. By letter dated June 19, 2018, the licensee provided a revised LLB analysis considering a break flow of 444 gpm to conservatively bound all possible configurations of the letdown system as well as any potential backflow through the break.

The NRC staff reviewed this response and determined that it is consistent with Wolf Creek's design and it conservatively captures the LLB and any back flow from the downstream letdown line piping. The time required for an operator to identify the accident and close the letdown isolation valve is expected to be within 30 minutes, 10 seconds after accident initiation. The licensee assumes based on the temperature and pressure conditions of the letdown line flow that 18 percent of the leaking coolant flashes to steam. In a letter dated December 4, 2017, the NRC staff asked the licensee to explain if the flashing fraction was determined consistent with RG 1.183 Appendix A, Regulatory Position 5.4 which uses a constant enthalpy process based on the maximum time-dependent temperature of the water circulating outside the containment. In its letter dated January 15, 2018, in response to RAI ARCB1-LLBA-1, the licensee stated:[24]

> The flashing fraction was calculated using the equation from Regulatory Guide 1.183, Appendix A, Position 5.4. The 18% flashing fraction was calculated using the letdown pressure of 600 psig and letdown temperature of 380 °F....

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[24] The quote is a combination of statements from Attachment I to the letter and Attachment I of Enclosure I.

> ...While 290 °F is the design temperature of the system, the value had been increased well above the design temperature to 380 °F for the analysis.
>
> However, while developing the responses to this question... it was determined that it is more appropriate to use a temperature of 290 °F as it is consistent with... the design conditions of the system... Therefore, the temperature value used for calculating enthalpy has been changed to 290 °F.
>
> ...This results in a calculated flashing fraction of 0.08.
>
> The updated analysis conservatively applied a 10% airborne fraction consistent with Regulatory Guide 1.183, Appendix A, Position 5.5.

The NRC staff reviewed this response and determined that it is consistent with Wolf Creek's design and that a 10 percent flashing fraction is consistent with RG 1.183, Appendix A, Regulatory Position 5.5. The iodine and alkali metal in the steam is assumed to become airborne and is available for release to the atmosphere, and all noble gases contained in the leaking primary coolant are available for release to the atmosphere. For the LLB, the licensee assumes the line break occurs outside of containment and radionuclides are directly released into the auxiliary building. The activity is assumed to be instantaneously transported to the Unit Vent Stack with no credit taken for dilution, holdup or cleanup in the auxiliary building. The licensee does not take any credit for the auxiliary building ventilation system filtration.

3.5.6.7.3 *Control Room Habitability for the LLB*

The licensee determined that the control room intake radiation monitor signal setpoint is immediately reached in the LLB analysis and the licensee credits the control room ventilation system switchover to emergency mode operation. As a result, the control room ventilation isolation is modeled in the dose analyses in two parts: actuation of the emergency mode filtration in both the control building and the control room, and closure of the HVAC intake damper. The actuation of the emergency mode filtration occurs following receipt of the high radiation signal. In the analyses, emergency mode filtration is actuated after a delay of at least 120 seconds to account for instrumentation delays and damper movement following an automatic isolation signal. The total unfiltered inleakage modeled during emergency mode is 50 cfm. Prior to the closure of the normal HVAC intake damper, the 50-cfm unfiltered inleakage is associated with the normal HVAC intake atmospheric dispersion factor. After closure of the normal HVAC intake damper, the unfiltered inleakage is apportioned between the emergency mode HVAC intake (40 cfm) and the communications corridor intake (10 cfm associated with ingress/egress).

During normal plant operation, unfiltered air from the environment enters the control room at a flow rate of 2,195 cfm and enters the control building at a flow rate of 14,360 cfm. Unfiltered inleakage to the control room during normal operation is 50 cfm. The atmospheric dispersion factor values for this accident are those of the unit vent stack. The control room and control building ventilations flow paths and flows are in Figure 3 and Table 4 of Attachment I to the licensee's supplement dated June 19, 2018 (Reference 10).

3.5.6.7.4 TSC Habitability for the LLB

The TSC ventilation system remains in normal operation for the duration of the LLB. During normal plant operation, unfiltered air from the environment enters the TSC at a flow rate of 550 cfm. Unfiltered inleakage during normal operation is 20 cfm. The licensee did not take credit for manual initiation of the TSC ventilation system emergency mode.

3.5.6.7.5 Conclusion for the LLB

The licensee evaluated the radiological consequences resulting from the postulated LLB and concluded that the radiological consequences at the control room and TSC are within the dose guidelines provided in 10 CFR 50.67 and the EAB and LPZ are within a small fraction (i.e., 10 percent) of the dose guidelines provided in 10 CFR 50.67. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.2 of this SE. The licensee's calculated dose results are given in Table 3.5.7-1. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds the licensee's analysis demonstrates with reasonable assurance that: (1) an individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem TEDE, (2) an individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release, during the entire period of its passage, would not receive a radiation dose in excess of 25 rem TEDE, and (3) adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. Therefore, the EAB, LPZ, control room, and TSC doses estimated by the licensee for the LLB were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.5.6.8 Waste Gas Decay Tank (WGDT) Leak or Failure

The WGDT accident considered is a rupture of one WGDT that results in the uncontrolled release of the entire inventory of the tank to the environment from the radwaste [radioactive waste] building over a 2-hour period. The accident is further described in UFSAR Section 15.7.1.5 (Reference 20). The licensee stated that in the proposed revised analysis, RG 1.183 models are applied to this analysis in conjunction with event guidance from RG 1.24 (Reference 27) and SRP, Branch Technical Position 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure" (Reference 18.l). A list of some of the input parameters and assumptions is provided in the "AST" column of Enclosure IV, Table 4.3-13, "Assumptions Used for Waste Gas Decay Tank Failure Analysis," and the noble gas activity associated with the gas tank is provided in Table 4.3-2a, "Tank Activities – AST," of the same enclosure. The WGDT iodine inventory are a factor of 100 higher than those provided in Table 4.3-2a because of the response to ARCB1-WT-5 in Attachment I to the letter dated June 19, 2018. The supplemental response for ARCB1-WT-5 in the letter dated December 6, 2018, revised the WGDT source term provided in Tables 4.2-4, "Waste Gas Decay Tank Activity," and 4.3-2a of Enclosure IV.

The NRC staff relied upon SRP, Branch Technical Position 11-5, which is referenced in SRP, Section 11.3 (Reference 18.f), and Wolf Creek's CLB to conduct a review of potential releases following the postulated failure of a WGDT and to determine the radiological consequences. As stated in Enclosure IV (see Section 4.3.10.3, "Acceptance Criteria") the EAB and LPZ dose

acceptance criterion for a WGDT leak or failure is 0.1 rem TEDE, and this criterion is consistent with SRP, Branch Technical Position 11-5, and the dose to an individual member of the public as described in 10 CFR Part 20, "Standards for Protection Against Radiation," limit (also discussed in RIS 2006-04 (Reference 35)). The control room dose limit is set forth in 10 CFR 50.67 as 5 rem TEDE.

The licensee assumes no control room isolation and that the control room ventilation remains in normal operation mode. The normal mode ventilation flow into the control building is assumed to be 14,360 cfm and the unfiltered inflow into the control room is assumed to be 2,195 cfm (see the supplementary response to RAI ARCB1-GENERAL-2 provided in Attachment I to the letter dated June 19, 2018).

The iodine is assumed to be 100 percent elemental. This is acceptable to the NRC staff because this assumption has no impact on the calculation results (no removal process is modeled, and the control room filters are not credited in the analysis).

3.5.6.8.1 Conclusion for the WGDT Leak or Failure

The licensee evaluated the radiological consequences resulting from the postulated WGDT leak or failure and concluded that the radiological consequences at the EAB, LPZ, control room and TSC are within the acceptance criteria found in the regulatory guidance identified in Section 2.2 of this SE. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.2 of this SE. The licensee's calculated dose results are given in Table 3.5.7-1 of this SE. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff's calculations confirmed the licensee's dose results and therefore, the EAB, LPZ, control room, and TSC doses estimated by the licensee for the WGDT leak or failure were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.5.6.9 Radwaste Liquid Waste Tank (LWT) Leak or Failure

The Radwaste LWT accident considered is defined as the uncontrolled atmospheric release from a boron recycle holdup tank or a hypothetical tank containing a maximized quantity of iodine as described in UFSAR Section 15.7.2.5 (Reference 20). The licensee stated that RG 1.183 models are applied to the revised analysis in conjunction with the analysis of record presented in UFSAR Section 15.7.2.5. Both tanks are analyzed to determine the bounding LWT rupture from a dose perspective. A list of input parameters and assumptions is provided in the "AST" column of Table 4.3-14 of Enclosure IV and the iodine and noble gas activity associated with the gas tank is provided in Table 4.3-2a of same enclosure.

The NRC staff relied upon WCNOC's licensing basis detailed in UFSAR 15.7.2.5 for this review. As stated in Enclosure IV (see Section 4.3.11.3, "Acceptance Criteria") the EAB and LPZ dose acceptance criterion for a LWT leak or failure is 0.1 rem TEDE, and this criterion is consistent with the dose to an individual member of the public as described in 10 CFR Part 20, limit. The control room dose limit is set forth in 10 CFR 50.67 as 5 rem TEDE.

The licensee assumes no control room isolation and that the control room ventilation remains in normal operation mode. The normal mode ventilation flow into the control building is assumed to be 14,360 cfm and the unfiltered inflow into the control room is assumed to be 2,195 cfm (see

the supplementary response to RAI ARCB1-GENERAL-2 provided in Attachment I to the letter dated June 19, 2018).

The iodine is assumed to be 100 percent elemental. This is acceptable to the NRC staff because this assumption has no impact on the calculation results (no removal processes are modeled, and the control room filters are not credited in the analysis).

3.5.6.9.1 Conclusion for the LWT Leak or Failure

The licensee evaluated the radiological consequences resulting from the postulated LWT leak or failure and concluded that the radiological consequences at the EAB, LPZ, control room, and TSC are within the acceptance criteria found in the regulatory guidance identified in Section 2.2 of this SE and as discussed above. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.2 of this SE or as provided in the licensee's CLB. The licensee's calculated dose results are given in Table 3.5.7-1 of this SE. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff's calculations confirmed the licensee's dose results and therefore, the EAB, LPZ, control room, and TSC doses estimated by the licensee for the LWT leak or failure were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.5.6.10 Loss of Non-Emergency AC Power (LOAC)

An LOAC event trips the RCPs and decreases forced flow through the reactor core. No fuel cladding damage or fuel melting is assumed to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed SG tube leakage, RCS activity passes from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through the ARVs and/or MSSVs. In addition, iodine and alkali metal activity is contained in the secondary coolant before the accident and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident.

3.5.6.10.1 Source Term for the LOAC

The analysis of the LOAC event is not discussed in RG 1.183. However, the licensee determined that the release pathway for this analysis is similar to the locked rotor event and the accident-initiated iodine spike is similar to the MSLB event. In the application, the licensee stated that the release pathway models are consistent with RG 1.183, Appendix G and accident-initiated iodine spiking models are consistent with RG 1.183, Appendix E. RG 1.183, Appendix E, Regulatory Position 2, states that "[i]f no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specifications," including the effects of pre-accident and concurrent iodine spiking. As stated above, the licensee's evaluation indicates that no fuel damage would occur as a result of an LOAC event. However, in the licensee's application the NRC staff could not find the discussion or analysis of the pre-accident iodine spike for the LOAC event. Therefore, in its letter dated December 4, 2017 (Reference 53), the NRC staff requested that the license submit an analysis or a description of the LOAC radiological consequences analysis assuming a pre-accident iodine spike for the NRC staff's review. In the letter dated January 15, 2018 (Reference 7), the licensee stated that the LOAC analysis source term is not consistent with RG 1.183, Appendix E, Regulatory Position 2.1 and that RG 1.183 Appendix E is for a MSLB

accident and not for the LOAC. During the regulatory audit conducted on March 19 and 20, 2018, the NRC staff explained that Wolf Creek TSs allow operation with a pre-accident iodine spike level and that a technical basis is needed for the current TSs or the TSs should be revise consistent with the new proposed LOAC analysis. The licensee acknowledged the TSs allowed operation with a pre-accident spike level and by its letter dated June 19, 2018, provided the revised LOAC analysis including a pre-accident iodine spike.

Therefore, the licensee considered the two radioiodine spiking cases as described in RG 1.183 Appendix E. The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated LOAC that has raised the primary coolant iodine concentration to the maximum value permitted by the TSs for a spiking condition. For Wolf Creek, the maximum iodine concentration allowed by TS LCO 3.4.16 as the result of an iodine spike is 60 microcuries per gram of DE I-131.

The licensee assumes that the primary system transient associated with the LOAC causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike. The increase in primary coolant iodine concentration for the concurrent iodine spike case is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the TS limit for normal operation. For Wolf Creek the RCS TS LCO 3.4.16 limit for normal operation is 1.0 microcurie per gram of DE I-131. The duration of the concurrent iodine spike is assumed to be 8 hours in accordance with RG 1.183.

For additional conservatism, the alkali metal activity in the RCS at the time of the LOAC event is at a 1 percent fuel defect level and the activity in the secondary coolant is assumed to be in the same ratio as the primary-to-secondary iodine concentrations. In addition, the noble gas activity concentration in the RCS at the time of the LOAC event is based on the Wolf Creek TS LCO 3.4.16 as 500 microcuries per gram of DE XE-133.

For the LOAC event, the licensee evaluated the radiological dose contribution from the release of secondary-side activity using the equilibrium secondary-side specific activity found in Wolf Creek TS LCO 3.7.18 as 0.1 microcuries per gram of DE I-131. The licensee assumes that the chemical form of iodine released from the SGs to the environment is 97 percent elemental and 3 percent organic consistent with RG 1.183, Appendix E, Regulatory Position 4.

3.5.6.10.2 Release Transport for the LOAC

The licensee followed the guidance as described in RG 1.183, Appendix G, Regulatory Position 5 in all aspects of the transport analysis for the LOAC event. The licensee assumes a total primary-to-secondary leak rate equal to 1 gpm, which is higher than the TS 3.4.13.d total allowable leak rate of 150 gpd primary-to-secondary leakage through any one SG.

RG 1.183, Appendix G, Regulatory Position 5.2, states:

> The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lb_m/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lb_m/ft^3).

The licensee's total primary-to-secondary leak rate of 1 gpm corresponds to a leakage density of 62.4 lb_m/ft^3. RG 1.183, Appendix G, Regulatory Position 5.3, states:

> The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F). The release of radioactivity should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

In the LOAC event, the condenser is assumed to be unavailable and the plant is cooled down by releasing steam to the environment via the MSSVs and ARVs. In accordance with RG 1.183, the licensee assumes that steam releases continue for 12 hours, at which time shutdown cooling is initiated using the RHR system.

In accordance with RG 1.183, the licensee assumes that all noble gas radionuclides released from the primary system are released through the SG to the environment without reduction or mitigation. Following the guidance from RG 1.183, Appendix G Regulatory Position 5.6, which directs the use of the assumptions in Regulatory Position 5.5 and 5.6 of RG 1.183, Appendix E, the licensee assumes that all of the primary-to-secondary leakage into the SG will mix with the secondary water without flashing during periods of total tube submergence. RG 1.183, Appendix E, Regulatory Position 5.5.4, states:

> The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

Accordingly, the licensee assumes that the radioactivity in the initial bulk water of the SG becomes vapor at a rate that is a function of the steaming rate and the inverse of the partition coefficient. The licensee used a partition coefficient of 100 for iodine released from the SG. The plant-specific moisture carryover factor of 0.25 percent to the steam releases is applied to the release of alkali metals from the secondary side.

For the LOAC, the primary-to-secondary leakage to intact SGs transport path leads to two release points at the MSSVs/ARV vents and the turbine-driven auxiliary feed water exhaust vent.

3.5.6.10.3 Control Room Habitability for the LOAC

The licensee determined that the control room intake radiation monitor signal setpoint is not reached in the LOAC analysis. As a result, the control room ventilation system remains in normal mode operation for the duration of the LOAC event and the licensee does not credit control room pressurization or any other safety functions of the control room ventilation system in the LOAC analysis. In normal mode, unfiltered air from the environment enters the control building and control room. During normal plant operation, unfiltered air from the environment enters the control room at a flow rate of 2,195 cfm and enters the control building at a flow rate of 14,360 cfm. Unfiltered inleakage to the control room during normal operation is 50 cfm and is included in the control room flow rate of 2,195 cfm. The atmospheric dispersion factor values used in the licensee's analysis for this release path are assumed as the larger value of the two release points.

3.5.6.10.4 TSC Habitability for the LOAC

The TSC ventilation system remains in normal operation for the duration of the LOAC event. During normal plant operation, unfiltered air from the environment enters the TSC at a flow rate of 550 cfm. Unfiltered inleakage during normal operation is 20 cfm. The licensee did not take credit for manual initiation of the TSC ventilation system emergency mode.

3.5.6.10.5 Conclusion for the LOAC

The licensee evaluated the radiological consequences resulting from the postulated LOAC and concluded that the radiological consequences at the control room and TSC are within the dose guidelines provided in 10 CFR 50.67 and the EAB and LPZ are within a small fraction (i.e., 10 percent) of the dose guidelines provided in 10 CFR 50.67. The NRC staff's review finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.2 of this SE. The licensee's calculated dose results are given in Table 3.5.7-1 of this SE. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the licensee's analysis demonstrates with reasonable assurance that: (1) an individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total TEDE, (2) an individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release, during the entire period of its passage, would not receive a radiation dose in excess of 25 rem TEDE, and (3) adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. Therefore, the EAB, LPZ, control room and TSC doses estimated by the licensee for the LOAC were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.5.6.11 Additional Design Basis Analysis Assumptions

Upon completion of the dose calculations proposed to support this LAR, the licensee initially added conservatism to the calculated dose values to arrive at the proposed dose result values. This additional dose "adder" increased the calculated doses by typically 10 percent. In a supplemental response for RAI ARCB1-GENERAL-3, dated December 6, 2018, the licensee stated that the doses no longer contain this "adder." Therefore, the NRC did not review or approve the use of the "adder" to the dose analyses reviewed in this SE.

3.5.6.12 Additional Supplemental Information

In several of the supplemental RAI responses, the licensee provided additional information describing interactions with the NRC staff. For example, in the supplemental response to RAI ARCB-FHA-5 and ARCB1-FHA-6, dated December 6, 2018, the licensee describes interactions with the NRC staff during the regulatory audit conducted on March 19 and 20, 2018, and during a meeting with the Technical Specifications Task Force. This information was not considered in the NRC staff's SE.

3.5.7 Evaluation of the Proposed Technical Specifications Changes Related to Adoption of Alternative Source Term

3.5.7.1 Changes to TS 1.1, "Definitions"

DOSE EQUIVALENT (DE) I-131

The NRC staff has evaluated the proposed definition of DOSE EQUIVALENT I-131 and has determined that the proposed definition, which uses the committed effective dose equivalent DCFs from Federal Guidance Report (FGR) No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988 (Reference 48), is acceptable because this definition will allow the licensee to calculate DE I-131 using the same DCFs as are used in the dose consequence analyses.

DOSE EQUIVALENT (DE) XE-133

The NRC staff has evaluated the proposed definition of DE XE-133 and has determined that the use of the effective DCFs from FGR No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," September 1993 (Reference 49), in the DE XE-133 definition is acceptable because they are consistent with the DCFs used in the dose consequence analyses.

3.5.7.2 Changes to LCOs 3.3.7 and 3.7.10 Regarding CREVS during Core Alterations

During the review of the analysis for the dropping of loads allowed over irradiated fuel assemblies, by letter dated December 4, 2017, the NRC staff through RAI ARCB-FHA-3 expressed a concern that the control room HVAC system was allowed to be inoperable during core alterations. The potential exists for an unirradiated fuel assembly (new fuel) to be dropped onto an irradiated assembly during core alterations. This would result in an FHA scenario in which the control room HVAC could not be credited. Therefore, it was requested that either an analysis be performed evaluating the resulting control room dose for the limiting event during core alterations, or that core alterations be added to the limits of applicability for the affected LCOs to preclude the event.

In order to address this concern, by letter dated June 19, 2018 (Reference 10), the licensee provided TS changes adding "during core alterations" to the limits of applicability for LCO 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation," (via added note (c) to Table 3.3.7-1) and LCO 3.7.10, "Control Room Emergency Ventilation System (CREVS)." The TS changes expanded the applicability of the LCO to include "during core alterations" via addition of new note (c) to Table 3.3.7-1 applicable to Functions 1, 2 and 3 and to the LCO applicability statement of LCO 3.7.10. Additionally, the words "or during core alterations" is added to Condition E of LCO 3.3.7 and Conditions D and E of LCO 3.7.10.

The NRC staff reviewed the proposed changes and determined that the changes would alleviate the concern by requiring the necessary instrumentation for CREVS actuation and the CREVS system itself to be operable during performance of core alterations. Additionally, by adding the explicit language to the proposed conditions in the actions table for both LCOs, the proper remedial actions of placing CREVS in CRVIS mode, suspending core alterations, and any movement of irradiated fuel assemblies would be taken to alleviate the concern. Therefore, the revised LCO applicability and actions collectively provide reasonable assurance that the health and safety of the public will not be endangered.

3.5.7.3 Changes to LCO 3.4.16 Regarding DE XE-133 ≥ 500 micro-ci/gm

The NRC staff noted that the LAR did not present an analysis assuming DE XE-133 was equal to or greater than 500 microcuries per gram (micro-ci/gm) at the start of the event for the review of the revised radiological consequence analysis for SGTR, MSLB and other accidents that assume DE XE-133. To address this concern the licensee presented a revised LCO 3.4.16 similar to a precedent approved for the Donald C. Cook Nuclear Plant (Cook), Unit Nos. 1 and 2, in Amendment Nos. 332 and 314, respectively (Reference 55). The LAR proposes to delete existing Condition B, revise and renumber Condition C as Condition B and carry forward Required Actions C.1 and C.2 as B.1 and B.2, respectively.

The revised Condition B includes required action and associated completion time of Condition A not met or DE X-133 not within limit or DE I-131 greater than 60 micro-curies per gram. The revised Required Actions require shutting down the plant by being in Mode 3 within 6 hours and Mode 5 within 36 hours.

As discussed in Sections 3.5.6.1, 3.5.6.2, 3.5.6.4, 3.5.6.7, and 3.5.6.10, the NRC staff reviewed the proposed changes and determined that the proposed Condition B and the revised Required Actions are conservative because they would require the licensee to shut down the plant for any of the included conditions. Therefore, the TS actions, as revised, collectively provide reasonable assurance that the health and safety of the public will not be endangered.

3.5.7.4 Changes to LCO 3.7.13 Regarding Movement of Fuel in the Fuel Building

During the review of the revised radiological consequence analysis for an FHA occurring in the fuel building, the NRC staff noted that Wolf Creek LCO 3.7.13, would permit the fuel building boundary to be open under administrative controls and Condition E would permit two EES trains to be inoperable (due to an inoperable boundary) for 24 hours during the movement of irradiated fuel in the fuel building. To address the concern regarding the allowance to continue moving fuel while the boundary is inoperable, the licensee proposed to revise the required action to require that movement of irradiated fuel assemblies in the fuel building immediately be suspended. The licensee also proposed deletion of Condition F.

The NRC staff reviewed the proposed changes and determined that the proposed Condition E and the deletion of Condition F would ensure that movement of irradiated fuel in the fuel building is suspended if two trains of EES are inoperable. As discussed in Section 3.5.6.2 of this SE, the TS requirement is aligned with the dose consequence analysis assumption in this regard. Therefore, the TS actions as revised will collectively provide reasonable assurance that the health and safety of the public will not be endangered.

3.5.7.5 Changes to LCOs 3.7.10, 3.7.13 and 3.9.4 Regarding Administrative Control of Boundaries

In the review of the proposed revised radiological consequence analyses, the NRC staff noted that existing TSs permitted the CRE, the CBE, the auxiliary building, and the fuel building boundaries to be open under administrative control. The TSs also permitted penetration flow paths providing direct access from the containment atmosphere to the outside atmosphere to be opened under administrative controls during core alterations and during movement of irradiated fuel assemblies within containment. The licensee's proposed radiological analyses did not appear to include these allowed configurations as inputs and assumptions in the analysis. The

staff requested that the licensee submit revised radiological analyses or propose changes to the TS to resolve this apparent inconsistency.

In the letter dated January 15, 2018, the licensee proposed a revision to the LCO 3.7.10 Note, LCO 3.7.13 Note and LCO 3.7.13 Condition B. The licensee proposed that the LCO Notes be clarified to state that the boundaries may be opened intermittently under administrative controls that ensure the boundary can be closed consistent with the safety analysis. The addition of this provision would ensure the assumptions in the accident analysis would be satisfied in the unlikely event that an accident occurred while a boundary was open under administrative controls.

In the letter dated June 19, 2018, the licensee proposed a similar revision to the LCO 3.9.4 Note. The revision would clarify that penetration flow paths may be unisolated under administrative controls that ensure the building boundary can be closed consistent with the safety analysis.

The NRC staff reviewed the proposed actions in the revised note and for Condition B of LCO 3.7.13 in Sections 3.5.6.1 and 3.5.6.2 of this SE and determined that the actions are sufficient and appropriate for the situation of having two EES trains inoperable due to an inoperable auxiliary building boundary. When the auxiliary building boundary is inoperable, action must be taken to lessen the effects on control room occupants and ensure radiological exposures will not exceed the calculated dose limits. The staff reviewed the proposed actions in the note revisions and for Condition B of LCO 3.7.13 and determined that the actions are sufficient and appropriate for the situation of having two EES trains inoperable due to an inoperable auxiliary building boundary. When the auxiliary building boundary is inoperable, action must be taken to lessen the effects on control room occupants and ensure radiological exposures will not exceed the calculated dose limits. The 24-hour Completion Time of Required Action B.2 is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions as directed by Required Action B.1. These actions are similar to those endorsed by the staff in NUREG-1431(Reference 19) as the appropriate required actions when one or more control room emergency ventilation trains inoperable due to an inoperable Control Envelope boundary while in Mode 1, 2, 3, or 4, requiring actions needed to protect the control room occupants. Therefore, the TS actions, as revised, will collectively provide reasonable assurance that the applicant will comply with the regulations and that the health and safety of the public will not be endangered.

3.5.7.6 Changes to TS 5.5.12

NRC RIS 2006-04 (Reference 35), Summary of Issue 11, "Acceptance Criteria for Off-Gas or Waste Gas System Release," states, in part:

> The acceptance criteria for this event is that associated with the dose to an individual member of the public as described in 10 CFR Part 20, "Standards for Protection Against Radiation." When the NRC revised 10 CFR Part 20 to incorporate a TEDE dose, the offsite dose to an individual member of the public was changed from 500 mrem whole body to 100 mrem TEDE. Therefore, any licensee who chooses to implement AST for an off-gas or waste gas system release should base its acceptance criteria on 100 mrem TEDE. Licensees may also choose not to implement AST for this accident and continue with their existing analysis and acceptance criteria of 500 mrem whole body.

The proposed change to TS 5.5.12 is to change the radiation exposure limit to an individual in an unrestricted area from whole body exposure of ≥ 0.5 rem to ≥ 0.1 rem TEDE and is consistent with the guidance in RIS 2006-04 and acceptable as per the NRC staff evaluation in Sections 3.5.6.8 and 3.5.6.9 of this SE.

3.5.7.7 Change to TS 5.5.18

The proposed change to TS 5.5.18, "Control Room Envelope Habitability Program," replaces the radiation exposure limit of 5 rem whole body or its equivalent to any part of the body for the duration of the accident for access and occupancy of the control room envelope under design basis accident conditions to 5 rem TEDE. The change is consistent with the limits specified in 10 CFR 20.1201, "Occupational dose limits for adults," and evaluated by the NRC staff for various DBAs in Section 3.5 of this SE, and therefore, is acceptable.

3.5.8 Summary of NRC Staff Conclusions for AST

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed license amendment at Wolf Creek. The NRC staff finds the analysis methods and assumptions consistent with the regulatory requirements and guidance specified in Section 2.2 of this SE. The staff compared the doses estimated by Wolf Creek to the applicable acceptance criteria and the results estimated by the staff in its confirmatory calculations. The staff finds with reasonable assurance that the licensee's estimates of TEDE due to DBAs will comply with the following requirements of 10 CFR 50.67: (1) an individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem TEDE, (2) an individual located at any point on the outer boundary of the LPZ, who is exposed to the radioactive cloud resulting from the postulated fission product release, during the entire period of its passage, would not receive a radiation dose in excess of 25 rem TEDE, and (3) adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident, and the guidance of RG-1.183. Therefore, the EAB, LPZ, control room and TSC doses estimated by the licensee and were found to meet the applicable accident dose criteria and are acceptable.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the Wolf Creek design basis is superseded by the AST proposed by WCNOC. The previous offsite and control room accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR Part 50.67 or fractions thereof, as defined in Regulatory Position 4.4 of RG 1.183 and as discussed in this SE. All future radiological accident analyses done to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined in the Wolf Creek design basis and modified by the present amendment.

Table 3.5.7-1
Total Effective Dose Equivalent per Accident in Roentgen Equivalent Man (rem)

Accident	EAB	LPZ	Regulatory Limit	Control Room	TSC	Regulatory Limit
Loss-of-Coolant Accident[25]	9.0	14.0	25	3.7	4.3	5
Fuel Handling Accident	1.2	0.39	6.3	1.1	1.2	5
Fuel Handling Accident – Auxiliary Building Releases	N/A	N/A	6.3	3.1	N/A	5
Main Steam Line Break						
Pre-accident Iodine Spike	0.20	0.12	25	4.5	0.28	5
Accident Initiated Iodine Spike	0.58	0.54	2.5	4.8	0.44	5
Steam Generator Tube Rupture						
Pre-accident Iodine Spike	0.99	0.32	25	4.2	2.2	5
Accident Initiated Iodine Spike	0.80	0.26	2.5	1.1	1.5	5
Locked Rotor Accident	0.38	0.32	2.5	3.5	0.16	5
Control Rod Ejection Accident						
Containment Release	1.1	1.9	6.3	2.6	2.0	5
Secondary Release	0.38	0.32	6.3	3.5	0.16	5
Letdown Line Break						
Pre-accident Iodine Spike	0.57	0.19	2.5	1.5	0.78	5
Accident Initiated Iodine Spike	0.35	0.12	25	0.37	0.43	5
Waste Gas Decay Tank Failure	0.090	0.029	0.1	0.057	0.0076	5
Recycle Holdup Tank Rupture	0.025	0.0080	0.1	0.053	0.0058	5
Liquid Waste Tank Failure	0.045	0.015	0.1	0.23	0.024	5
LOAC						
Pre-accident Iodine Spike	0.0018	0.0015	2.5	0.86	0.0021	5
Accident Initiated Iodine Spike	0.0013	0.0047	25	2.5	0.0034	5

3.6 Transition to Westinghouse Core Design and Safety Analyses

The licensee plans to replace the existing Wolf Creek methodology for performing core design non-LOCA, and LOCA safety analyses (for post-LOCA subcriticality and long-term core cooling only) with the standard Westinghouse developed and NRC-approved analyses methodologies.

The results of the NRC staff evaluations are discussed in the following sections.

3.6.1 Thermal Conductivity Degradation

Correctly modeling fuel behavior is an important part of transient and accident analysis. In Information Notice (IN) 2009-23, "Nuclear Fuel Thermal Conductivity Degradation" (Reference 56), and its supplements, the NRC notified the stakeholders about the impact of irradiation on fuel thermal conductivity degradation (TCD). This IN complements the information previously provided by the NRC regarding fuel thermal performance analysis codes that do not account for fuel TCD.

[25] The EAB and LPZ doses for the LOCA represent the limiting case discussed in Section 3.5.6.1.3.3 of this SE.

IN 2009-23 informed the stakeholders, including Westinghouse, that thermal performance codes approved by NRC before 1999 did not include this reduction in thermal conductivity with increasing irradiation because earlier test data were inconclusive as to the significance of the effect. Any codes used for safety analyses that incorporate data starting at the fuel rod level and generated by the pre-1999 models may mischaracterize the expected plant performance.

The fuel performance code that was used to perform the Wolf Creek analyses to support this methodology transition is one of the fuel performance codes that does not account for TCD. The NRC staff review of the non-LOCA analyses, specifically the MSLB and RCCA ejection analyses determined that further evaluation was required to quantify the impact of TCD and available margins to ensure there are appropriate constraints on fuel system operation and demonstrate that the plant satisfies GDC 10 of Appendix A to 10 CFR Part 50 requirements.

By letter dated October 4, 2018 (Reference 57), the NRC staff issued an RAI asking the licensee to demonstrate that, among others, the MSLB accident and RCCA ejection accident analyses appropriately include the effects of nuclear fuel TCD in the evaluation against the acceptance criteria as discussed in Enclosure I (Reference 1).

The licensee reanalyzed MSLB and RCCA events using burnup dependent thermal conductivity information from NRC-approved WCAP-17642-P-A, Revision 1, "Westinghouse Performance Analysis and Design Model (PAD5),"[26] November 2017 (Reference 58), to address the issue of TCD and provided the results of the analysis by letter dated March 5, 2019 (Reference 15). This approach addresses the thermal conductivity input assumptions used in the analyses discussed in this LAR. The impacts of this change on the MSLB and RCCA ejection accident events are discussed below in Sections 3.6.3 and 3.6.4. Prior NRC staff review experience has indicated that the MSLB and RCCA ejection accident can be significantly impacted by TCD, whereas the remaining, non-LOCA transients are minimally impacted, based on the way those events are modeled in the Westinghouse methodology.

3.6.2 Summary of Non-LOCA Events

Wolf Creek plans to transition from its current methodologies for performing core design, non-LOCA and post-LOCA subcriticality, and long-term core cooling to the Westinghouse methodologies for performing these analyses. WCAP-17658-NP, Revision 1, "Wolf Creek Generating Station Transition of Methods for Core Design and Safety Analyses – Licensing Report," was provided as Enclosure I of the LAR dated January 17, 2017.

The licensee stated in the LAR that for this transition method, the safety analyses that were reanalyzed at the higher nominal power level associated with a measurement uncertainty recapture (MUR) power uprate. The reanalysis effort did not assume any other plant or analysis input changes that may be required to support an actual MUR power uprate. Also, the core design effort did not assume any other plant or analysis input changes that may be required to support an actual MUR power uprate. Even though some analyses were performed at an uprated power (representative of an MUR), the MUR conditions (i.e., nuclear steam supply system power) would be bounding for plant operation at current RTP. The licensee made it clear that the intent of this licensing amendment application is not to request approval of a MUR power uprate.

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[26] Referred to as PAD5, thereafter.

The acceptance criteria for all the non-LOCA accident analyses are described in Enclosure I and are summarized in Section 2.1 of this SE. The NRC staff reviewed the information provided in the LAR and its attachments and concluded that non-LOCA analyses that are reanalyzed as part of this LAR meet the acceptance criteria listed in the Wolf Creek UFSAR Chapter 15 and meet the guidance in NUREG-0800 related to specific non-LOCA analysis.

The following Table presents a list of all the non-LOCA transient events that were considered by the licensee in support of the transition method program for Wolf Creek including cross references to the applicable UFSAR sections. It is noted that for the transient event that is not reanalyzed, an evaluation had been performed by the licensee that considered the available information and existing plant specific calculations to assess the event acceptability.

Non-LOCA Transient Events Analyzed or Evaluated			
Transient Event	**Report Section**	**UFSAR Section**	**Analyzed or Evaluated?**
Feedwater system malfunctions that result in a decrease in feedwater temperature	2.2.1	15.1.1	Analyzed
Feedwater system malfunctions that result in an increase in feedwater flow	2.2.2	15.1.2	Analyzed
Excessive increase in secondary steam flow	2.2.3	15.1.3	Analyzed
Inadvertent opening of a SG atmospheric relief or safety valve	2.2.4	15.1.4	Analyzed
Steam system piping failure (steam line break (SLB)) at zero power	2.2.5.1	15.1.5	Analyzed
SLB at full power	2.2.5.2	15.1.5	Analyzed
Loss of external electrical load, turbine trip, inadvertent closure of main steam isolation valves, and loss of condenser vacuum	2.3.1	15.2.2 15.2.3 15.2.4 15.2.5	Analyzed
Loss of non-emergency AC power to the station auxiliaries	2.3.2	15.2.6	Analyzed
Loss of normal feedwater flow	2.3.3	15.2.7	Analyzed
Feedwater system pipe break	2.3.4	15.2.8	Analyzed
Partial loss of forced reactor coolant flow	2.4.1	15.3.1	Analyzed
Complete loss of forced reactor coolant flow	2.4.1	15.3.2	Analyzed
RCP shaft seizure (locked rotor) and RCP shaft break	2.4.2	15.3.3 15.3.4	Analyzed
Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	2.5.1	15.4.1	Analyzed
Uncontrolled RCCA bank withdrawal at power	2.5.2	15.4.2	Analyzed
RCCA misoperation (dropped RCCA, dropped RCCA bank, statically misaligned RCCA, single RCCA withdrawal)	2.5.3	15.4.3	Analyzed
Startup of an inactive RCP at an incorrect temperature	2.5.4	15.4.4	Evaluated

Non-LOCA Transient Events Analyzed or Evaluated			
CVCS malfunction that results in a decrease in the boron concentration in the reactor coolant (boron dilution)	2.5.5	15.4.6	Analyzed
Spectrum of RCCA ejection accidents	2.5.6	15.4.8	Analyzed
Inadvertent operation of the ECCS during power operation	2.6.1	15.5.1	Analyzed
CVCS malfunction that increases reactor coolant inventory	2.6.2	15.5.2	Analyzed
Inadvertent opening of a pressurizer safety or relief valve	2.7.1	15.6.1	Analyzed
Anticipated Transient Without Scram	2.8	15.8	Analyzed

The following Table provides a roadmap of the Westinghouse analysis codes used and identifies the applicable UFSAR Section(s) associated with the use of each code.

Subject	Topical Report	Code(s)/ Method(s)	UFSAR Section
Non-LOCA Thermal Transients	WCAP-7908-A (Reference 59))	FACTRAN	15.4.1, 15.4.8
Non-LOCA Safety Analysis	WCAP-14882-P-A (Reference 60)	RETRAN	15.1.1, 15.1.2, 15.1.3, 15.1.4, 15.1.5, 15.2.2, 15.2.3, 15.2.4, 15.2.5, 15.2.6, 15.2.7, 15.2.8, 15.3.1, 15.3.2, 15.3.3, 15.3.4, 15.4.2, 15.5.1, 15.5.2, 15.6.1
	WCAP-7907-P-A (Reference 61)	LOFTRAN	15.4.2, 15.4.3, 15.8
Non-LOCA Thermal/ Hydraulics	WCAP-11397-P-A (Reference 62)	RTDP	Thermal Hydraulic (T/H) Design
	WCAP-14565-P-A (Reference 63)	VIPRE	15.1.2, 15.1.4, 15.1.5, 15.3.1, 15.3.2, 15.3.3, 15.3.4, 15.4.1, 15.4.2, 15.4.3
Neutron Kinetics	WCAP-7979-P-A (Reference 64)	TWINKLE	15.4.1, 15.4.8
Multi-dimensional Neutronics	WCAP-10965-A (Reference 65)	ANC	15.1.2, 15.1.4, 15.1.5, 15.4.3
Steam Generator Tube Rupture	WCAP-10698-P-A (Reference 66)	RETRAN	15.6.3
	WCAP-14882-P-A (Reference 60)		

Table 2.1-6, "Non-LOCA Results Summary," of Enclosure I summarizes the results obtained for each of the non-LOCA transient analyses.

The input assumptions for these analyses are documented in Enclosure I. The NRC staff has not audited the reanalyzed calculations, or the input assumptions used. However, the staff has reviewed and determined that the information provided are consistent with the Wolf Creek TSs and are acceptable. In addition, the licensee has confirmed that the input assumptions in Enclosure I are chosen in a conservative manner in comparison to their original analysis.

It is noted that although the analyses and evaluations were performed with the intent to make them cycle-independent, the reload safety evaluation process described in WCAP-9272, (Reference 22) will be applied for future fuel reloads to verify that reload-related safety analysis inputs remain bounding.

In Enclosure I, the licensee indicated that Wolf Creek has reviewed the analyses related to the effects of the proposed methodology transition on the thermal/hydraulic (T/H) design of the core and the RCS. The licensee concluded that the analyses demonstrated that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to proven designs, (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermos-hydrodynamic instability. In its letter dated March 5, 2019 (Reference 15), the licensee further demonstrated that all non-LOCA events will continue to meet the applicable NRC regulatory requirements of GDC 27 and GDC 35 of Appendix A to 10 CFR Part 50 following implementation of the proposed methodology transition.

The NRC staff reviewed the results for all of the non-LOCA analyses including the MSLB and RCCA ejection, which are specifically addressed in Sections 3.6.3 and 3.6.4 of this SE and determined that the results demonstrate that all applicable safety analyses acceptance criteria are satisfied for Wolf Creek.

3.6.2.1 Conclusion

The NRC staff reviewed the non-LOCA analyses results included in this LAR and its enclosures, specifically, the DNBR limits and margin summary listed in Table 2.12-3 provided in Enclosure I. In Enclosure I of the LAR, the licensee stated that the non-LOCA analyses results meet all the applicable acceptance criteria. The staff review determined that the analyses performed by the licensee for the methodology transition continue to meet all non-LOCA events, including MSLB and RCCA, as discussed in more detail in Sections 3.6.3 and 3.6.4.

3.6.3 Main Steam Line Break Accident Evaluation

The MSLB event is described in Section 15.1.5 of the Wolf Creek UFSAR. The MSLB involves the rupture of a main steam line, which, because of the excessive increase in steam flow, results in an overcooling of the RCS and potential return to power event.

The acceptance criteria, with the potential to be most impacted by TCD for this accident, are related to satisfying the DNB design basis and precluding fuel centerline melting.

The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is possibility that the core will become critical and return to power. A return to power following a steam line rupture is a potential concern mainly because of the high-power peaking factors which exist, assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid solution delivered by the ECCS, and fuel centerline melting is precluded.

The proposed Wolf Creek licensing basis analysis for a MSLB, while the reactor is at power, has been performed to confirm the conclusions documented in Westinghouse Report WCAP-9226-P-A, Revision 1, " Reactor Core Response to Excessive Secondary Steam Releases," February 1988 (Reference 67) are valid, even with assumed parameters being outside the generic limits established by Westinghouse. This analysis, which does not address possible TCD impacts, demonstrates that the DNB design basis is satisfied, and fuel centerline melting is precluded.

The analyzed MSLB cases for Wolf Creek assume initial hot shutdown and hot full power (HFP) conditions at time zero. These cases are used to identify a limiting initial condition. At the time of an MSLB, when power level reaches a trip setpoint, the reactor will be tripped by the normal overpower protection system. Following a trip at power, the RCS contains more stored energy than at no load, the average coolant temperature is higher than at no load, and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no load conditions of RCS temperature and shutdown margin assumed in the analyses are reached.

The MSLB TR WCAP-9226-P-A Revision 1 was approved by the NRC on January 31, 1989 (Reference 68). The MSLB TR SE identifies two conditions of acceptance that are discussed below:

1. This condition is satisfied for the MSLB TR application to Wolf Creek since the computer codes used in the non-LOCA analyses in this LAR are Westinghouse WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999 (Reference 60), WCAP-10965-A, Revision 0, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1986 (Reference 65), and WCAP-14565 -P-A "VIPER-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal Hydraulic Safety Analysis," October 1995 (Reference 63) codes, which are all NRC accepted and approved computer codes.

2. For the pressure between 500 and 1000 pounds per square inch absolute (psia), the 95/95 departure from DNBR limit for the W-3 correlation is 1.45.

As discussed in Section 2.12, "Thermal and Hydraulic Design," of Enclosure I, the W-3 DNB correlation has been replaced with the Westinghouse low pressure (WLOP) DNB correlation, which has a different 95/95 DNBR limit. Table 2.1-6 (non-LOCA Results Summary) of Enclosure I presents the DNBR safety analysis limit applied in the MSLB analysis for which the WLOP DNB correlation was used. By letter dated October 4, 2018 (Reference 57), in the RAI related to the MSLB event, the NRC staff requested the licensee to demonstrate how the MSLB analysis results and acceptance criteria appropriately include the effects of TCD, including:

a) The hot zero power (HZP) and hot full power from nucleate boiling ration, and
b) The hot full peak linear heat rate

In its letter dated March 5, 2019, the licensee explained that the RETRAN code, which is the T/H transient analysis code, is an NRC approved methodology and was used for this MSLB analysis. Therefore, there are no changes to the at HZP and at HFP MSLB DNBR results due to the effects of TCD.

While the transient analyses of the system response for the HZP and HFP MSLB accident are not impacted by the effects of TCD, the power to melt limits are affected by TCD. Therefore, the power-to-melt limit used in the analysis of the HFP MSLB transient was recalculated by the licensee to explicitly consider the effects of TCD. The transient analysis methods, in combination with Wolf Creek cycle-specific core design parameters, were then used to recalculate the corresponding peak linear heat generation rate observed during the HFP MSLB event. The result for the limiting HFP MSLB case are summarized in letter dated March 5, 2019 (Reference 15), that shows that the peak fuel linear heat generation rate does not exceed the value that would cause fuel melting.

The calculation of the peak fuel linear heat generation rate is based upon the upcoming fuel cycle (Cycle 24). As the peak fuel linear heal generation rate can change on a cycle-specific basis, an additional check will be performed as part of the reload process to confirm that the power-to-melt limit with the effects of TCD considered continues to be met for the HFP SLB accident. Wolf Creek is planning to fully implement the PAD5 models and methods into its licensing basis for the non--LOCA analyses. Once the PAD5 models and methods are fully incorporated into the Wolf Creek licensing basis, the PAD5 analysis will become the analysis of record and the power-to-melt limit will be checked as part of standard reload process.

The results of the recent MSLB analysis performed, in response to the RAI transmitted by letter dated October 4, 2018, demonstrates that the DNB design basis continues to be met with the effects of TCD explicitly included in the analysis. Additionally, the peak fuel linear heat generation rate does not exceed the value that would cause fuel center line melting.
The analysis of the MSLB event assumed input assumptions that bound the values for the upcoming fuel cycle 24.

The NRC staff determined that the licensee adequately addressed the effects of TCD relative to the MSLB because the licensee used NRC-approved methods (i.e., PAD5, to determine the power-to-melt limits and performed a demonstration calculation to model the event in a way that explicitly accounts for TCD effects). This calculation showed that fuel melting would not occur. The licensee also stated that the core design information used in the analysis would be confirmed on a cycle-specific basis. The NRC staff concluded that the licensee provided reasonable assurance that the capability to cool the core would be maintained because fuel melt would not be anticipated under postulated MSLB conditions, and that the limiting parameters used in the analysis would be checked each reload. Therefore, the NRC staff concludes that the Wolf Creek fuel system is taking TCD into account and is acceptable.

3.6.4 Rod-Cluster Control Assembly Ejection Accident Evaluation

The RCCA ejection accident is defined as the mechanical failure of a control rod driver mechanism pressure housing resulting in the ejection of the RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

The analysis of the RCCA ejection accident is performed in two steps as follows:

1. A spatial neutron kinetics computer code, Westinghouse WCAP-7979-P-A, "TWINKLE – A Multi-dimensional Neutron Kinetics Computer Code," dated January 1975 (TWINKLE), approved by the NRC staff on July 29, 1974 (Reference 64), that is used to calculate the core average nuclear power transient, including the various core feedback effects; that is, Doppler and moderator reactivity; and

2. The FACTRAN computer code, Westinghouse WCAP-7908-A, "FACTRAN – A FACTRAN IV Code for Thermal Transients in UO_2 Fuel Rod," December 1989, approved by the NRC staff by letter dated September 30, 1986 (FACTRAN) (Reference 59) used the average nuclear power calculated by TWINKLE and performs a fuel rod transient heat transfer calculation to determine the core average heat flux and hot spot fuel temperature transients.

The RCCA ejection accident is described in Section 15.4.8, "Spectrum of Rod Cluster Control Assembly Ejection Accidents," of the Wolf Creek UFSAR. This accident is defined as a mechanical failure of a control rod drive mechanism pressure housing resulting in the ejection of the RCCA and drive shaft. The consequence of this mechanical failure is a rapid, positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. The NUREG-0800 guideline for this event is to ensure the resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by the reactor trip actuated by high nuclear power signals that ensure damage to the reactor coolant pressure boundary is limited to local yielding, and the core in a coolable geometry.

From the NRC-approved analysis methodology, the criterion applied to ensure the core remains in a coolable geometry following a rod ejection incident, is that the average fuel pellet enthalpy at the hot spot must remain less than 200 calories per gram (cal/gm) (360 British thermal units per pound mass (Btu/lb_m). The use of the initial conditions presented in Table 2.5.6-1, "Selected Input and Results of the Limiting RCCA Ejection Analyses," of Enclosure I resulted in conservative calculations of the fuel pellet enthalpy. The results of the licensing basis analyses without consideration of TCD, demonstrated that the fuel pellet enthalpy does not exceed 360 Btu/lb_m for any of the rod ejection cases analyzed. Overpressurization of the RCS during a rod ejection event is generically addressed in Westinghouse WCAP-7588, Revision 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," dated January 1975 (Reference 70).

Another applicable acceptance criterion from the NRC-approved analysis methodology is that fuel melting must be limited to less than the innermost 10 percent of the fuel pellet at the hot spot, even if the average fuel pellet enthalpy at the hot spot is less than the limit of 360 Btu/lb_m. As stated in the letter dated March 5, 2019, conservative fuel melt temperatures were assumed for the hot spot for the beginning of life and end of life cases, respectively. These fuel melting temperatures correspond to a specific burnup limit at the hot spot. The peak UO_2 burnup at the hot spot is based on the assembly with the maximum post-ejection total peaking factor, which is typically a fresh fuel assembly. Therefore, the fuel melting temperatures represent bounding values based on the assumed UO_2 burnup at the hot spot. The maximum burnup at the hot spot at beginning of life and end of life is confirmed to be below these values as part of the reload process. This assumption does not affect the maximum licensed fuel burnup limit. The results of the licensing basis rod ejection analyses demonstrated that the amount of fuel melting was limited to less than 10 percent of the fuel pellet at the hot spot for each of the rod ejection cases.

In Enclosure I, Section 2.1.13, "Results Summary," the licensee states that "Table 2.1-6 summarizes the results obtained for each of the non-LOCA transient analyses." The NRC staff reviewed the analysis results provided in Enclosure I, specifically the RCCA ejection accident, and concluded that the analysis results demonstrates that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits are not exceeded.

In Wolf Creek TS 2.1.1.2, the peak centerline temperature shall be maintained ≤ 5080 °F, decreasing by 58 °F per 10,000 MWD/MTU of burnup. In the reanalyzed RCCA ejection event, the licensee had conservatively adjusted the fuel melting temperature to correspond to TS 2.1.1.2 burnup dependent safety limit for all analyzed cases (Reference 15). As expected, with the effects of TCD included, the peak fuel average and centerline temperature increased (Table 1, Reference 15).

By letter dated March 5, 2019, in its response to the RAI that requested the licensee disposition TCD for the RCCA ejection accident, the licensee reanalyzed the RCCA. Based on the results provided in its RAI response, the licensee had shown that all the acceptance criteria for the RCCA ejection event continue to be met for Wolf Creek's upcoming Cycle 24.

Since the licensee addressed the effects of TCD on the RCCA ejection accident using NRC-approved methods, and showed that the acceptance criteria remained satisfied, the NRC staff concluded that the Wolf Creek fuel system design would comply with the requirements of GDC 28 of Appendix A to 10 CFR Part 50. Therefore, the NRC staff determined that the RCCA ejection accident, revised with PAD5-based fuel data, demonstrated acceptable results. Therefore, the NRC staff concludes that the Wolf Creek fuel system is taking TCD into account in its non-LOCA analyses.

3.6.5 Post-LOCA Subcriticality and Long-Term Core Cooling

3.6.5.1 Post-LOCA Subcriticality

The new UFSAR Chapter 15 transient and accident analyses are included in WCAP-17658-NP, Revision 1-C, included as Enclosure I to the licensee's submittal dated January 17, 2017.

The licensee indicated that the methodology for demonstrating compliance with the requirements of 10 CFR 50.46(b) is documented in part in Westinghouse WCAP-8339, (Reference 21) and states that borated water injected by the ECCS will maintain the core subcritical following a LOCA. To support demonstrating compliance with this position, in Section 2.7.4.1 of WCAP-17658-NP, a curve for post-LOCA containment sump boron concentration, as a function of the pre-LOCA RCS boron concentration, was developed.

The curve assumes uniform mixing in the sump from all significant water sources, including the RCS, the RWST (the primary suction source for pumped safety injection), and the accumulators. Sources of high boron concentration are assumed to have minimum mass and boron concentration. All sources are assumed to spill directly to the sump with no holdup elsewhere in containment.

The NRC staff reviewed the assumed volumes and boron concentrations for the various water sources and found them to be appropriately bounding based on comparison to the plant TSs and UFSAR. Given that the source for pumped safety injection will not switch to the sump until the RWST has been emptied, and that until that point, the water in the core will mostly be highly

borated water from the ECCS, the assumption of uniform mixing from all sources represents the most dilute state that would be experienced in the reactor following a LOCA and is thus appropriately bounding.

The sump boron concentration is calculated as a function of the pre-LOCA RCS boron concentration, assuming peak xenon. Westinghouse proposed that the way to verify post-LOCA subcriticality on a cycle-specific basis would be to ensure that the maximum critical boron concentration for the cycle (for an all-rods-out, no xenon, condition between 68 and 212 °F) falls below the curve. The NRC staff finds this acceptable because it will ensure that post-LOCA subcriticality will be maintained throughout the cycle.

Thus, the NRC staff finds that the method for calculating the post-LOCA sump boron concentration is acceptable and uses acceptably conservative inputs to generate a post-LOCA sump boron concentration curve as a function of the pre-LOCA RCS conditions. The use of this curve to determine acceptability of a core design relative to the post-LOCA subcriticality criterion is also found to be acceptable.

3.6.5.2 Post-LOCA Long Term Core Cooling

Post-LOCA long term core cooling (LTCC) evaluations consist of both a BAP analysis and a decay heat removal analysis. If boric acid precipitates out of solution in the core, core cooling may be impeded. Therefore, a BAP analysis is used to demonstrate that the maximum boric acid concentration in the core remains below the boric acid solubility limit. The decay heat removal analysis is used to demonstrate that the decay heat in the fuel can be removed in the long term and that the core remains covered.

PWR licensees, including Wolf Creek, typically prevent BAP by switching at least a portion of the ECCS injection from the cold leg to the hot leg, providing flushing flow that reduces the boric acid concentration in the core. An analysis is performed to predict the amount of time following a LOCA before the incipient BAP point is reached. The result is used to inform operator actions to realign plant systems for hot leg injection. The CLB hot leg switchover time at Wolf Creek, documented in Sections 6.3.2.5 and 15.6.5.2 of the Wolf Creek UFSAR, is 10 hours.

The licensee's post-LOCA LTCC evaluation was performed by Westinghouse and is documented in Section 2.7.4.2 of WCAP-17658-NP. The following sections of this SE will document the NRC staff's review of the licensee's proposed BAP analysis methodology, decay heat removal analysis methodology, and their respective results.

3.6.5.3 Boric Acid Precipitation Analysis Methodology

In a letter dated August 1, 2005 (Reference 71), the NRC staff withdrew approval for Westinghouse's evaluation model for evaluating post-LOCA LTCC, which was documented in Combustion Engineering TR CENPD-254-P, "Post-LOCA Long-Term Cooling Evaluation Model," June 1980 (Reference 72). Though the NRC staff suspended approval of the TR, the staff also observed that the overall framework and general approach of CENPD-254-P remain valid for the purposes of what is to be included in the post-LOCA LTCC analysis. The NRC staff notes that the licensee did not request the use of CENPD-254-P but used a similar approach that the staff considers to be acceptable provided several key issues are addressed.

The following issues were identified for post-LOCA LTCC analyses in a letter from the NRC staff to Westinghouse dated November 23, 2005 (Reference 73).

- The assumed mixing volume should be justified and should account for voiding.
- The mixing volume should be time-variant and should account for loop pressure drop.
- The boric acid solubility limit should be justified, especially if elevated containment pressures or containment sump additives are considered in its determination.
- Evaluation models, based on 10 CFR Part 50 Appendix K, "ECCS Evaluation Models," should use a 1.2 multiplier on the decay heat.

In Section 2.7.4.2 of WCAP-17658-NP, Westinghouse stated that the LTCC evaluation model used for Wolf Creek is consistent with the "interim methodology" reported in an August 23, 2006, meeting between the NRC staff and the PWR Owner's Group (Reference 74). It was stated that the "interim approach" would be consistent with that was adopted in the Beaver Valley Power Station extended power uprate. The approach used was discussed in Sections 5.2.3 and 5.2.4 of the, "Beaver Valley Power Station Extended Power Uprate Licensing Report," (Reference 75) and found acceptable in the NRC staff's SE related to Beaver Valley Power Station extended power uprate, dated July 19, 2006 (Reference 76).

The LTCC evaluation provided by Westinghouse in Section 2.7.4.2 of WCAP-17658-NP, provided a disposition to several of the issues discussed in the NRC staff's August and November 2005 letters (References 71 and 73) to Westinghouse regarding post-LOCA LTCC analyses. The time-dependent boric acid concentration in the core region was computed considering the effects of core voiding on the mixing volume. For both large and small break LOCAs, a boric acid solubility limit at atmospheric conditions was assumed. Though a slight increase in the boiling point of water to 218 °F was assumed due to the presence of boric acid in the solution, the effects of containment sump additives on the BAP limit were not credited. The decay heat used was the 1971 ANS decay heat standard with a 1.2 multiplier. All these assumptions are consistent with the NRC staff's position from the letter dated November 23, 2005 (Reference 73) and are thus acceptable.

However, the referenced methodology, combined with the additional detail contained in the Wolf Creek documentation, did not provide sufficient information for the NRC staff to determine the acceptability of the models used to determine the transient mixing volume, including consideration of voids. In RAI-1 by e-mail dated September 21, 2017 (Reference 77), the NRC staff asked the licensee for additional information about LTCC evaluation model. The NRC staff requested for documentation and further details on the void model, RCS loop pressure drop calculations, mixing volume assumptions, and the calculation done to verify that sufficient decay heat will be removed after the switch to cold leg recirculation (and the subsequent switch to hot leg recirculation). In its response to RAI-1, by letter dated October 18, 2017 (Reference 5), the licensee provided the following response.

The licensee in its response to RAI-1.a stated that the post-LOCA core boron concentration calculations were performed with Version 9.0 of Westinghouse's SKBOR code, "A Computer Code for Calculating the Accumulation of Boric Acid in the Reactor Vessel" (SKBOR) (See Attachment 2 to Enclosure I of the letter dated October 18,2017, Reference 5) with documentation in Attachment 2 of letter dated October 18, 2017. SKBOR performs a simplified

mass conservation calculation, where coolant is stored in and transferred between two control volumes representing the RCS and containment sump. Borated liquid water is injected from the sump into the core at a rate that matches the boiloff rate (because any coolant added in excess of the boiloff rate is assumed to spill back out the break). Coolant boils off in the core, exiting the "core" control volume as saturated steam containing no boric acid, and is assumed to condense in containment and return to the sump as a saturated liquid. The assumption that no boric acid is entrained in the vaporized coolant is conservative with respect to the timing of BAP. The NRC staff also expects that conditions in the containment atmosphere will quickly reach saturation following a LOCA, and the assumption that the containment is already at saturation (and that all coolant exiting the core will condense into the sump) is reasonable and therefore acceptable. Overall, though SKBOR is a highly simplified representation of post-LOCA conditions in the RCS and containment, the approach is reasonable provided that the assumptions regarding the volume of water available in the core for mixing, containment and RCS pressures, and boric acid solubility limit are conservative. The NRC staff finds the approach used in SKBOR to be acceptable. The particular modeling assumptions made by the licensee for the Wolf Creek SKBOR calculations will be discussed later in this section.

SKBOR assumes an axially and radially uniform power distribution. However, the time-dependent void fraction is calculated as an axially-dependent parameter and is then averaged over the core volume. As discussed in the licensee's response to RAI-1.b, the Cunningham-Yeh model is used to calculate the axially-dependent void fraction in the core volume. This model has been well validated in the scientific literature for post-LOCA conditions, as discussed in the RAI-1 response, and the NRC staff finds it to be acceptable for this application.

As discussed in the licensee's response to the NRC staff RAI-1.d, the assumed volume available for mixing includes the core region, the upper plenum from the top of the core to the bottom of the hot leg, and the lower plenum. The effective mixing volume in the core and upper plenum is reduced by the calculated void fraction. The core exit void fraction is applied in the upper plenum, conservatively reducing the mixing volume. Also, the lower plenum volume available for mixing is reduced by 50 percent to account for the boric acid concentration gradient that inhibits effective mixing in the region. This 50 percent reduction is the same as that found in the NRC-approved SE for the Waterford Steam Electric Station, Unit 3 amendment for extended power uprate (Reference 78); therefore, the NRC staff finds this to be acceptable in this application, along with the other mixing volume assumptions.

The RCS loop pressure drop following the LOCA could potentially affect the mixing volume by suppressing the two-phase mixture level in the core. However, the effects of loop pressure drop on the core level were not directly considered in the SKBOR calculation. Instead, as discussed in the licensee's response to RAI-1.a, the loop differential pressure is evaluated in a post-processing step to ensure that there is sufficient margin in the calculated mixing volume to account for loop pressure drop effects. The calculation ensures that the static head of the liquid in the downcomer minus the static head of the liquid in the core is greater than or equal to the pressure drop in the loop. The static head is calculated by determining the collapsed liquid levels in the core and downcomer, accounting for voiding in both regions and increased fluid density in the core from the boric acid concentration. The loop pressure drop is calculated using the Darcy formula based on the boiloff rate, with a hydraulic loss coefficient that includes the effects of a locked reactor coolant pump rotor. The NRC staff finds this approach to be acceptable, since it will ensure that there is sufficient driving head between the downcomer and the core to overcome the loop pressure drop. The assumptions used to calculate the collapsed

liquid levels and pressure drops are appropriate for the scenario being modeled and are therefore, acceptable for the application to Wolf Creek.

Following a LOCA, it is possible that steam could condense in the broken RCS loop and collect in the low point of the loop, known as the loop seal, near the RCP suction. This could interfere with the transit of steam to and out the break, resulting in a buildup of pressure (and concurrent decrease in two-phase mixture level) in the core region. As discussed in Section 2.7.4.2.1.4, "Results," of WCAP-17658-NP, Westinghouse did not include this potential loop seal blockage in the pressure drop model because Rig-of-Safety Assessment tests and LOCA evaluation model calculations have not predicted sustained loop seal blockage and instead predicted cyclic loop seal clearing and refilling. Westinghouse stated that this cyclic refilling and clearing would result in enhanced mixing between the downcomer, lower plenum, and core volume, increasing the effective mixing volume and lowering the core boric acid concentration. Thus, Westinghouse considers it conservative to not model loop blockage. Based on the Westinghouse's assessment, the NRC staff determined that there is a reasonable assurance that neglecting loop seal blockage and clearing, the LTCC analyses using SKBOR is a conservative assumption, and therefore, finds it to be acceptable.

As discussed in Section 2.7.4.2.1.3. "Description of Analyses and Evaluations," of WCAP-17658-NP, the earliest entry to hot leg recirculation is determined by subtracting 1 hour from the latest acceptable hot leg recirculation time, as defined by the BAP time. The adequacy of the decay heat removal ability is checked at the earliest time to hot leg recirculation. The response to RAI-1.a states that the earliest time to hot leg recirculation is also confirmed against the liquid film entrainment threshold in the RCS hot legs, to ensure effective ECCS recirculation flow after the transfer to hot leg recirculation. The liquid film entrainment threshold is evaluated using the Wallis-Steen liquid entrainment onset criterion and the Ishii-Grolmes inception criterion. Gas mass flow rates at the inception of liquid film entrainment are calculated for each criterion and related to the core decay heat through the core steaming rate. This relationship to decay heat is used to determine the latest time following shutdown to drop below the entrainment threshold, which is subsequently compared to the earliest time to begin the transfer to hot leg recirculation to ensure that the ECCS recirculation flow to the hot legs will be effective. The use of Wallis-Steen and Ishii-Grolmes criteria include several conservative assumptions (e.g., reduced dimensionless gas velocity for onset of entrainment and assumption of rough turbulent liquid film flow, respectively), and have been accepted by the NRC staff in other applications. Therefore, the NRC staff finds their use to be acceptable.

The adequacy of dilution flow is calculated as a post-processing step, as discussed in the response to RAI-1.a. Coolant from the sump, with the corresponding boric acid concentration, is added to the core at the rate of injection provided by the hot leg recirculation flow (which exceeds the boiloff rate). Rather than allowing the dilute coolant injected from the hot leg to displace the concentrated solution in the core, the post-processing calculation assumes that the dilution flow expands the mixing volume, resulting in a slower dilution rate. Because it is done as a post-processing step, the increased mixing volume does not affect the boiloff rate or any other parameter included in the main calculation, just the resulting boric acid concentration. The NRC staff finds this approach to be acceptable, since it adequately represents the important phenomena in the dilution process.

3.6.5.4 Boric Acid Precipitation Analysis Evaluation

Westinghouse performed calculations for both large break LOCA (LBLOCA) and small break LOCA (SBLOCA), in the cold leg. Hot leg breaks were not considered in this application because there is a constant supply of flushing flow in excess of the boiloff rate, which is consistent with previous post-LOCA LTCC evaluations and is acceptable to the NRC staff. The LBLOCA analysis was performed assuming saturated conditions in the core at atmospheric pressure, while the SBLOCA analysis was performed assuming saturated conditions in the core at 120 psia.[27] As previously discussed, both calculations assumed boric acid solubility at atmospheric pressure conditions, which is conservative, especially for the SBLOCA case. During cold leg recirculation, one RHR pump, one intermediate head safety injection (IHSI) pump, and one centrifugal charging pump (CCP) inject coolant from the sump into the RCS. However, because the break is in the cold leg, the downcomer can only fill up to the bottom of the cold leg, and any ECCS injection, in excess of the boiloff rate, spills into containment through the break. During hot leg recirculation, one IHSI pump is realigned to provide flushing flow to two hot legs, while the RHR pump and CCP continue inject to all four cold legs. These flow conditions were found by the NRC staff to represent the limiting post-LOCA ECCS configurations and therefore, are acceptable. Other inputs, including the specific core power, coolant volumes, and boric acid concentrations, were documented in Table 2.7.4-2 of WCAP-17658-NP. The NRC staff notes that none of the parameters listed in this table are changing because of this LAR. The NRC staff reviewed the parameter values provided in Table 2.7.4-2 of WCAP-17658-NP against the values provided in the Wolf Creek TS and UFSAR and found that the analysis inputs were the same or conservative relative to the plant values.

The latest acceptable time to complete the transfer to hot leg recirculation was found to be 7.5 hours from the initiation of the accident. The SBLOCA was limiting regarding BAP control, though both the SBLOCA and LBLOCA analyses still show some margin to the boric acid solubility limit at 7.5 hours. After flushing flow is initiated, both cases show a rapid decrease in boric acid concentration. Based on the methodology discussed above, the earliest entry to hot leg recirculation is established as 6.5 hours, which was checked against the entrainment threshold and found to be acceptable. The NRC staff determined that a window of 6.5 to 7.5 hours, following a LOCA for operators to complete a switch to simultaneous injection, is acceptable based on the analysis presented by the licensee.

An additional evaluation was performed for a condition where hot leg switchover was not established until 12 hours, to demonstrate the effectiveness of dilution flow when the RCS remains at pressure for an extended period. At 12 hours, this case simulated a 100 °F/hour cooldown for 1 hour, coincident with switchover of ECCS injection to the hot leg. Unlike the regular SBLOCA and LBLOCA cases, this case assumed the boric acid solubility limit at the saturation temperature corresponding to the system pressure (which decreased from 120 psia to slightly above atmospheric during the cooldown). The evaluation demonstrated that there is substantial margin between the core boric acid concentration and the incipient BAP limit, both at pressure and throughout the cooldown, providing an indication of the conservatism of the methodology used by Wolf Creek. Therefore, the NRC staff found that there is reasonable assurance that BAP will be avoided after a 100 °F/hour cooldown following a SBLOCA.

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[27] This was not explicitly stated in the licensee's submittal but was verified by the NRC staff by comparing the core boiloff rates between the small and large break LOCA cases.

3.6.5.5 Decay Heat Removal Analysis Methodology and Results

The licensee stated in the response to RAI-1.e that the method for verifying sufficient decay heat removal during cold leg recirculation was documented in Westinghouse Energy Systems NSAL 95-001, "Minimum Cold Leg Recirculation Flow," dated January 12, 1995 (Reference 79). This letter recommends that the minimum ECCS flow during cold leg recirculation meet or exceed 1.2 times the boiloff rate from decay heat at the time cold leg recirculation is initiated. The response to RAI-1.e also details the method for verifying sufficient decay heat removal during hot leg recirculation, which the licensee stated that it was documented in NSAL-92-010, "Hot Leg Switchover Methodology" (Reference 80) and NSAL-95-001. As discussed in the RAI response, these letters recommend that, for simultaneous hot and cold leg recirculation plants such as Wolf Creek, the hot legs should receive 1.3 times the decay heat boiloff rate in ECCS injection flow while also maintaining an injection rate of 1.2 times the decay heat boiloff rate to the cold legs at the time hot leg recirculation is initiated. However, the staff notes that requiring flow substantially, in excess of a conservatively calculated boiloff rate, would be expected to provide conservative results. Section 2.7.4.2.1.3 of WCAP-17658-NP describes the conditions used to evaluate the time of transfer to cold leg recirculation, which include maximum ECCS injection from two RHR pumps, two IHSI pumps, two CCPs, and two containment spray pumps. This maximizes the ECCS flow, thus maximizing the drain-down rate of the RWST and minimizing the time of entry to cold leg recirculation. Once cold leg recirculation is entered, decay heat removal checks are performed assuming minimum flows based on the failure of a single emergency diesel generator resulting in one operable ECCS train consisting of a single RHR pump, one IHSI pump, and one CCP. As previously discussed, the hot leg recirculation flow configuration assumes one RHR pump and one CCP injecting to all four cold legs and one IHSI pump injecting to two of the four hot legs. The cold leg and hot leg recirculation pump configurations represent the configurations that provide the minimum expected ECCS flow, considering the single worst failure in the system, and therefore, are acceptable.

The licensee stated that the minimum flow requirements discussed in the response to RAI-1.e were met at both the transfer to cold leg recirculation and the transfer to hot leg recirculation. The NRC staff reviewed the Wolf Creek ECCS design as documented in Chapter 6 of the Wolf Creek UFSAR and found that even a single train of ECCS is capable of providing makeup flow far in excess of the boiloff rate reported in Figures 2.7.4-3 and 2.7.4-4 of WCAP-17658-NP (which is, in and of itself, conservatively calculated with a 1.2 multiplier on the ANS 1971 decay heat curve). As such, the NRC staff finds that Wolf Creek will have sufficient decay heat removal capability at the transfer to cold leg recirculation and the transfer to hot leg recirculation, based on the minimum flow condition described in Section 2.7.4.2.1.3 of WCAP-17658-NP.

3.6.5.6 Generic Safety Issue-191

None of the analyses included in the licensee's submittal address Generic Safety Issue (GSI)-191, which concerns the effects of post-LOCA debris on PWR licensees' capability to maintain adequate long-term cooling of the reactor core. As discussed in the letter dated May 16, 2013, from WCNOC, "Docket No. 50-482: Wolf Creek Nuclear Operating Corporation Proposed Path to Closure of Generic Safety Issue-191, 'Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance'" (Reference 81), the licensee plans to pursue a risk-informed approach to close GSI-191 at Wolf Creek. Because the transition to Westinghouse safety analysis methodologies and AST radiological consequence analysis methodology will not have an impact on the Wolf Creek GSI-191 resolution path, the NRC staff does not consider it necessary for the current licensing action to address this issue.

3.6.5.7 Conclusions Regarding Post-LOCA LTCC Analyses

The NRC staff finds that the licensee has documented adequate post-LOCA long term core cooling capability. The licensee has established a BAP control plan that will allow the operators a period of 1 hour to realign systems to establish simultaneous injection to provide flushing flow and prevent BAP. Additionally, the licensee demonstrated that adequate decay heat removal will be provided by the ECCS system throughout the post-LOCA period, including at the transition to sump recirculation and at the transition to hot leg injection.

3.6.6 Evaluation of the Proposed Technical Specifications Changes Related to Transition to Westinghouse Core Design and Safety Analyses

The TS changes evaluated in this SE resulting from the transition to Westinghouse methodologies are described below.

3.6.6.1 TS Section 2.1.1, "Reactor Core SLs"

Current TS Section 2.1.1, "Reactor Core SLs [Safety Limits]," states, in part,

> "In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:
>
> 2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.23 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation."

The proposed change would revise SL 2.1.1.1 above as follows:

> 2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.13 for the ABB-NV DNB correlation, and ≥ 1.18 for the WLOP DNB correlation.

Wolf Creek is licensed to meet the requirements of GDC 10 of Appendix A to 10 CFR Part 50 which requires the reactor core and associated coolant, control, and protection systems be designed to assure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and AOOs. This is accomplished by having a DNB design basis which corresponds to a 95 percent probability at a 95 percent confidence level (the 95/95 DNB criterion) that DNB will not occur, and by requiring that fuel centerline temperature stays below melting temperature. The reactor core safety limits are established to prevent violation of these criteria.

SL 2.1.1.1 currently represents the DNBR design limit for the WRB-2 correlation. This design limit identifies the RTDP that is contained in WCAP-11397-P-A (Reference 62). The RTDP design limit DNBR is only applicable to accidents that are initiated at the HFP condition. The proposed new DNBR limits listed in the LAR for SL 2.1.1.1 reflect the NRC-approved DNBR correlation limit values for the WRB-2 correlation, ABB-NV correlation, and WLOP correlation which cover the DNB design bases for all accident analyses. As discussed in the SE for the

VIPRE-W Code (WCAP-14565-PA and its Addendum 2-P-A, (Reference 63)) the ABB-NV correlation is used for conditions to which WRB-2 is not applicable, such as the RCCA bank withdrawal from subcritical, and the WLOP correlation is used in DNB analysis of HZP of MSLB events.

The applicability of each of these NRC-approved methods to Wolf Creek is discussed in Section 3.6.6.5. This proposed change is based on the application of NRC-approved methodologies to Wolf Creek and therefore, the staff concludes that the proposed change to TS 2.1.1.1 is acceptable.

3.6.6.2 Addition of New TS 3.1.9, "RCS Boron Limitations < 500 °F"

The proposed change will add a new TS 3.1.9, "RCS Boron Limitation < 500 °F." The proposed change will revise the Applicability of Function 2.b., "Power Range Neutron Flux-Low," in Table 3.3.1-1 and add a new Applicability (Applicable Modes), new Conditions V, W, and X, and the appropriate SRs required to demonstrate Operability of the Function. Also, new Footnotes f, h, and i are added to address the revised Applicability of the Function.

This proposed change assures that the required mitigative capability is available in the form of adequate shutdown margin, or an automatic reactor trip for an RCCA bank withdrawal event that may be postulated to occur during low power or startup (subcritical) conditions. The new LCO will require that the RCS boron concentration shall be greater than ARO critical boron concentration when there is the potential for an inadvertent RCCA bank withdrawal due to a malfunction of control rod drive system or operator error. Westinghouse NSAL-00-016, "Rod Withdrawal from Subcritical Protection in Lower Modes" (Reference 82), in great details discusses this phenomenon and the reactor trip functions assumed in the uncontrolled RWFS condition event.

The uncontrolled RCCA bank withdrawal event for Wolf Creek is analyzed from both a subcritical and a low power startup condition (Wolf Creek UFSAR Chapter 15.4). This event is terminated by the Power Range Neutron Flux-Low Trip Function. The Power Range Neutron Flux-Low Trip Function is only capable of providing protection for RCCA bank withdrawal event when the RCS temperature is greater than or equal to 500 °F due to calibration issues associated with shielding caused by cold water in the downcomer region of the reactor vessel.

Since, the licensee has not performed explicit RCCA bank withdrawal analysis for MODES 3, 4, and 5, a new LCO 3.1.9 is added to ensure that the RCS will be borated to greater than the ARO critical boron concentration to provide sufficient shutdown margin in these MODES in the event of RCCA bank withdrawal event from a subcritical condition when the RCS temperature is below 500 °F.

The NRC staff review found that the proposed change is more restrictive than the existing WCGS TS due to additional requirements being added in the form of the LCO 3.1.9 on boration requirements when the RCS temperature is below 500 °F. Therefore, the staff concludes that the proposed change is consistent with the post-LOCA subcriticality evaluation conclusion discussed in Section 3.6.5 of this SE. Therefore, the NRC staff concludes that the proposed TS change is acceptable.

3.6.6.3 TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"

LCO 3.4.1 states,

> RCS DNB parameter for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limit specified below:
>
> a. Pressurizer pressure is greater than or equal to the limit specified in the COLR;
>
> b. RCS average temperature is less than or equal to the limit specified in the COLR; and
>
> c. RCS total flow rate ≥ 37.1×10^4 gpm and greater than or equal to the limit specified in the COLR.

The proposed change would revise the RCS total flow to 361,200 gpm. As described below, the total flow value has been previously incorporated into the licensing basis analysis by the licensee for Wolf Creek.

The non-LOCA safety analyses that are performed by the licensee, with the assumed total flow rate of 361,200 gpm in the T/H DNBR analyses are listed below:

- Feedwater system malfunctions that result in an increase in feedwater flow;
- Inadvertent opening of a SG atmospheric relief or safety valve;
- Steam system piping failure at zero power;
- Loss of external electrical load, turbine trip, inadvertent closure of main steam valves, and loss of condenser vacuum;
- Loss of non-emergency AC power to the station auxiliaries;
- Loss of normal feedwater flow;
- Feedwater system pipe break;
- RCP shaft seizure (locked rotor) and RCP shaft break (peak RCS pressure / peak clad temperature case);
- Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition;
- Uncontrolled RCCA bank withdrawal at power (peak RCS pressure cases); and
- Spectrum of RCCA ejection accidents.

In addition, the NRC SE for WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," dated January 1999 (Reference 83), states that if RCS flow rate were to be relocated to the COLR, the minimum limit for RCS total flow based on a staff approved analysis should be included in the TS. Therefore, the total flow value of 361,200 gpm, is the minimum RCS flow rate that is retained in the TS to assure that a lower flow rate than that approved by the NRC staff would not be used.

This proposed change is consistent with the total flow rate used in the present Wolf Creek UFSAR accident analyses and the reanalyzed non-LOCA input assumptions and is acceptable.

3.6.6.4 TS 3.7.1, "Main Steam Safety Valves (MSSVs)"

TS 3.7.1, "Main Steam Safety Valves (MSSVs)," LCO requires five OPERABLE MSSVs per SG. TS Table 3.7.1-1, "OPERABLE Main Steam Safety Valves versus Maximum Allowable Power," specifies the power limits (in percent RTP) applicable when the number of OPERABLE MSSVs per SG is less than five. Wolf Creek's UFSAR, Chapter 15 loss of load/turbine trip (LOL/TT) analysis assumes all MSSVs are OPERABLE. The table below specifies the present values listed in Wolf Creek TS Table 3.7.1-1 and are compared with the proposed LAR values:

Number of Operable MSSVs per SG	Maximum Allowable Power %RTP	
	Present Value	Proposed Value
4	87	70
3	65	51
2	44	31

A complete LOAC as described in Wolf Creek UFSAR Sections 15.2.2, "Loss of External Electrical Load"; 15.2.3, "Turbine Trip"; 15.2.4, "Inadvertent Closure of Main Steam Isolation Valves"; and 15.2.5, "Loss of Non-Emergency AC Power to the Station Auxiliaries (Blackout)" can result in a loss of power to the plant auxiliaries, such as RCP, condensate pumps, etc. The loss of power may be initiated by a complete loss of the offsite grid combined with a turbine generator trip, or by a loss of onsite AC distribution system.

Based on the expected frequency of occurrence, the LOAC is considered to be a Condition II event, and for Wolf Creek the LOAC event is bounded by an LOL/TT event.

The analyses that supports Wolf Creek's current UFSAR LOAC event is described in Section 2.3, "Decrease in Heat Removal by the Secondary System," of Enclosure I to the LAR. In addition, to the analyses that supports the UFSAR LOL/TT event, Wolf Creek performed a supplementary analysis of this event that supports operation at reduced power levels with one or more inoperable MSSVs. This supplementary analysis, which forms the basis for the values shown in the above TS Table 3.7.1-1, involved a more exhaustive analysis with an iterative process of running LOL/TT RETRAN cases for various power levels and moderator temperature coefficients with one, two, or three inoperable MSSV(s) per loop modeled.

The supplementary LOL/TT analysis determined the respective maximum initial power level for which the resultant peak main steam system pressure satisfied the acceptance DNB criteria and ensures that the 110 percent design pressure is not exceeded, as described in Wolf Creek's UFSAR and Enclosure I to the LAR.

In Attachment I to letter dated January 17, 2017, the licensee stated that calculations performed for this proposed change are consistent with those considered in the past, as documented in Enclosure I.

The NRC staff reviewed the licensing report prepared in support of the proposed change as documented in Enclosure I (WCAP-17658-NP), specifically Section 2.3, and concluded that the proposed change is consistent with the NRC regulations described in GDC 10 of Appendix A of 10 CFR Part 50, which requires the reactor core and associated coolant, control, and protection system be designed with appropriate margin so that the specified fuel design limits are not

exceeded during any condition of normal operation, and the non-LOCA events. For LOL/TT event, the licensee demonstrated that the DNB acceptance criteria and the RCS overpressure limit are met. Therefore, the fuel cladding and RCS integrity is maintained.

3.6.6.5 TS 5.6.5, "Core Operating Limits Report (COLR),"

TS 5.6.5, "Core Operating Limits Report," Section b, lists the analytical methods used to determine the core operating limits.

The proposed change would delete the following analytical methods listed in Section b of TS 5.6.5:

- WCNOC Topical Report TR 90-0025 W01, "Core Thermal Hydraulic Analysis Methodology for the Wolf Creek Generating Station."
- WCNOC Topical Report NSAG-006, "Transient Analysis Methodology for the Wolf Creek Generating Station."
- WCNOC Topical Report NSAG-007, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station."
- NRC Safety Evaluation Report dated March 30, 1993, for the "Revision to Technical Specification for Cycle 7."

The proposed change would add WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology" (Reference 22), to the analytical methods listed in TS 5.6.5.b. WCAP-9272-P-A is the only methodology that is associated with the determination of a TS COLR parameter.

WCAP-9272-P-A discusses a condition for using the RETRAN code for a Non-LOCA Safety Analysis. This condition requires selection of conservative input assumptions in the plant specific RETRAN model. In Section A.3, "RETRAN for Non-LOCA Safety Analysis," of Appendix of Enclosure I, the licensee states that,

> The input data used in the RETRAN analyses performed by Westinghouse came from both WCNOC and Westinghouse sources. Assurance that the RETRAN input data is conservative for the WCGS is provided via Westinghouse's use of transient-specific analysis guidance documents. ... safety analysis input values used in the WCGS analyses were selected to conservatively bound the values expected in subsequent operating cycles.

The NRC staff's review indicates that the proposed change adds the NRC-approved Westinghouse analytical methodology and removes the WCNOC analytical methodologies, which will no longer be used, to those listed in TS 5.6.5.b. This proposed change ensures adequate plant safe operation and is acceptable.

3.6.6.6 Conclusions Regarding the Proposed TSs Changes

The NRC staff finds that the licensee has provided adequate justification in support of the proposed TSs 2.1.1, 3.1.9, 3.4.1, 3.7.1, and 5.6.5. The licensee has established that Wolf Creek will operate safely while operating with the proposed TSs changes delineated in the LAR.

The proposed TSs changes are consistent with 10 CFR 50.36 providing reasonable assurance that the plant will continue safe operation and is acceptable.

3.6.7 AST Implementation.

3.6.7.1 Gap Release Fractions for AST

As part of the AST implementation, the licensee performed dose analyses for a number of transients and accidents. The dose analyses are documented in Enclosure IV of the LAR (Reference 1). Of the transients and accidents considered by the licensee, gap release was modeled for the locked rotor transient (Section 4.3.5 of Enclosure IV), rod ejection accident (Section 4.3.6 of Enclosure IV), LOCA (Section 4.3.9 of Enclosure IV), and the FHA (Section 4.3.12 of Enclosure IV). Tables 4.3-8, 4.3-9, 4.3-12, and 4.3-15 of Enclosure IV list the respective assumptions of these dose analyses and specifically note the assumed gap fractions.

As discussed above, RG 1.183 provides guidance on AST implementation. Footnote 10 of in Section 3.2 of RG 1.183 notes that the gap release fractions provided in Tables 1–3 are acceptable for LWR fuel with a maximum burnup of 62 GWD/MTU. Footnote 11 further notes that the non-LOCA gap fractions listed in Table 3 have been found acceptable for LWR fuel with peak burnup of 62 GWD/MTU, provided that the maximum linear heat generation rate (LHGR) does not exceed 6.3 kw/ft at burnups greater than 54 GWD/MTU.

3.6.7.1.1 Gap Fractions for Locked Rotor, Rod Ejection, and Loss-of-Coolant Accidents

The gap fractions provided in Table 4.3-8 of Enclosure IV for the locked rotor event are consistent with those of Table 3 of RG 1.183. The gap fractions assumed for the rod ejection accident are provided in Table 4.3-9 of Enclosure IV and are consistent with Table 3 and Footnote 11 of RG 1.183, which states that gap fractions for iodines and noble gases should be assumed at 10 percent for PWR rod ejection accidents. LOCA gap fractions provided in Table 4.3-12 of Enclosure IV are consistent with Table 2 of RG 1.183. As such, the NRC staff concludes that the gap fractions for all of these events are acceptable.

3.6.7.1.2 Gap Fractions for FHA

The licensee stated that the analysis for the FHA assumed that the Footnote 11 burnup and LHGR limits would not be met by 100 percent of the fuel. For fuel that does not meet the LHGR limits, Footnote 11 allows licensees to perform fission gas release calculations for the NRC staff to review on a case-by-case basis, provided that the calculations use a power history that bounds the limiting plant-specific power history.

The FHA analysis methodology employed by the licensee included assumed gap release fractions based on values reported in NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," dated February 1988 (Reference 84), and RG 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" (Reference 52). As discussed in Enclosure VII of the letter dated the January 17, 2017, the licensee considered the gap fractions provided in RG 1.25 and NUREG/CR-5009 to not be constrained by the Footnote 11 burnup limits in RG 1.183. The licensee believed that assumed gap fractions could be applied to higher burnups to bound expected Wolf Creek power histories. The licensee provided further clarification in its response to an RAI, by letter dated May 4, 2017 (Reference 3). In its response the licensee stated that the gap fractions from

RG 1.25 and NUREG/CR-5009 were considered to be applicable up to a peak power density of 20.5 kw/ft for average burnups exceeding 50 GWD/MTU, and that an overall rod-average burnup limit of 62 GWD/MTU is applicable to Wolf Creek.

In RG 1.25 it is stated that the assumptions were valid for the following peak assembly conditions: maximum LHGR of 20.5 kw/ft, maximum fuel centerline temperature of 4500 °F during operation, and assembly-average burnup of 25 GWD/MTU (corresponding to a peak local burnup of about 45 GWD/MTU).

The RG 1.25 gap release fractions are 30 percent of krypton-85, 10 percent of other noble gases, and 10 percent of radioactive iodines. Calculations from NUREG/CR-5009 were used to update these gap fractions for high burnup fuel and indicated that the I-131 gap fraction should be increased to 12 percent. The NUREG/CR-5009 gap release fractions were calculated for rod-average burnups of up to 60 GWD/MTU using the 1982 version of the ANS standard for gap release fraction calculations, ANS-5.4, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel." More specific operating history of the fuel was not provided.

RG 1.25 was withdrawn in 2016 because its guidance was updated and incorporated into RG 1.183 and RG 1.195 (Reference 85). The NRC staff evaluated the gap release fractions proposed by the licensee and based on RG 1.25 and NUREG/CR-5009 against the acceptable gap fractions discussed in RG 1.183 and PNNL-18212, Revision 1. In this particular application for Wolf Creek, the licensee's assumed 20.5 kw/ft LHGR limit for the applicability of the assumed gap release fractions approaches a value that could result in fuel centerline melt. As such, the NRC staff indicated its concern about the applicability of the NUREG/CR-5009 gap release fractions to this heat generation rate (let alone the ability of the fuel to withstand such a high LHGR, particularly at high burnup).

In its response dated June 19, 2018 (Reference 10), supplemented by letter dated August 9, 2018 (Reference 11) to RAI-2 issued via email dated September 21, 2017 (Reference 77), the licensee clarified that a maximum of 10 percent of fuel rods in the limiting assembly will be allowed to exceed the RG 1.183, Table 3, Footnote 11 gap fraction applicability limits. A check will be applied as part of the reload safety analysis checklist to ensure that this upper limit of 10 percent is met.

The LHGRs and burnups of the remaining 10 percent of the fuel that exceeds the RG 1.183, Table 3, Footnote 11 conditions will be bounded by calculations performed in Table 2.9 of PNNL-18212, Revision 1 (Reference 24). The power history assumed in PNNL-18212, Revision 1, Table 2.9 is 12.2 kw/ft up to 35 GWD/MTU, decreasing to 7.0 kw/ft at 65 GWD/MTU. The licensee stated that this will also be validated on a cycle-specific basis as part of the reload safety analysis checklist.

In the response to RAI-2, the licensee demonstrated the conservatism of the assumed gap fractions by showing that the gap release assumed in the FHA analysis bounds that expected for an assembly that met the power history limits assumed in the supplemental RAI responses. To do this, the licensee constructed a "composite" gap fraction assuming gap fractions from RG 1.183 for the 90 percent of rods in the assembly that meet the applicability limits, and the PNNL-18212, Revision 1, gap fractions for the 10 percent of the rods in an assembly fail the RG 1.183 Footnote 11 applicability limits but fall within the PNNL-18212, Revision 1, applicability limits. The gap fractions analyzed in the licensee's AST application were found to be more conservative than those in the "composite" assembly.

However, the NRC staff reviewed the CLB for Wolf Creek and found that it assumes the FHA damages one full assembly and 20 percent of an additional assembly. The NRC staff replicated the licensee's approach to generate a composite assembly but also accounts for the additional gap release from the second damaged assembly, increasing the overall gap release by 20 percent. However, for the composite gap fraction approach to be bounding, all rods that fail to meet the RG 1.183 Footnote 11 limits in the additional damaged assembly, must be assumed to fall within the 20 percent of rods that is expected to fail. As such, the NRC staff apportioned the release appropriately according to the assumption on fuel that fails to meet the Footnote 11 limits.[28] The staff finds that the licensee's gap fraction assumptions remain conservative even considering the 20 percent additional gap release from the second damaged assembly.

3.6.7.2 Conclusions Regarding Gap Release Fraction Assumptions for AST

The NRC staff accepts the licensee's use of the RG 1.25 and NUREG/CR-5009 gap fractions based on the demonstrations of conservatism relative to the RG 1.183 and PNNL-18212, Revision 1 gap fractions provided in the response to the staff's RAI-2 and its associated supplements, and evaluations performed by the NRC staff. The acceptance of the licensee's gap fraction assumptions is based on 90 percent of the fuel rods in the limiting assembly meeting the RG 1.183, Table 3, Footnote 11 applicability limit and the remaining 10 percent of the fuel rods meeting the PNNL-18212, Revision. 1, Table 2.9 applicability limits. As discussed, the licensee has incorporated reload checks to ensure that these applicability limits are met.

3.6.8 Conclusion

The NRC staff finds that the licensee's proposed transition to Westinghouse Core Design and Safety Analysis methodologies will utilize current approved methodologies and provide alignment with the Westinghouse fleet of plants. This transition is consistent with the regulations outlined in Section 2.1 of this SE.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Kansas State official was notified of the proposed issuance of the amendment on April 9, 2019. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration published in the *Federal Register* on October 2, 2018 (83 FR 49590), and there has been no public comment on such finding. Accordingly, the

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[28] The NRC staff's analysis assumed that 90 percent of fuel rods in the primary assembly are bounded by the RG 1.183 Footnote 11 gap fractions, and 10 percent are bounded by the PNNL-18212, Revision 1, Table 2.9 gap fractions. Of the 20 percent of fuel rods in the second assembly that are damaged, half of the fuel rods (or 10 percent of the overall assembly) were assumed to be bounded by the RG 1.183 Footnote 11 gap fractions and half were assumed to be bounded by the PNNL-18212, Revision 1, Table 2.9 gap fractions.

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) there is reasonable assurance that 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.

7.0 REFERENCES

1. McCoy, J. H., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: License Amendment Request for the Transition to Westinghouse Core Design and Safety Analyses," dated January 17, 2017 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML17054C103).

2. Reasoner, C. O., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Errata for the License Amendment Request for the Transition to Westinghouse Core Design and Safety Analyses," dated March 22, 2017 (ADAMS Accession No. ML17088A635).

3. McCoy, J. H., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Response to Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Transition to Westinghouse Core Design and Safety Analyses," dated May 4, 2017 (ADAMS Accession No. ML17130A915).

4. McCoy, J. H., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety Analyses," dated July 13, 2017 (ADAMS Accession No. ML17200C939).

5. McCoy, J. H., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety Analyses Including Adoption of Alternative Source Term," dated October 18, 2017 (ADAMS Accession No. ML17297A478).

6. McCoy, J. H., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety Analyses Including Adoption of Alternative Source Term," dated November 14, 2017 (ADAMS Accession No. ML17325A982).

7. Reasoner, C., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety Analysis Including Adoption of Alternative Source Term," dated January 15, 2018 (ADAMS Accession No. ML18024A477).

8. McCoy, J. H., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Supplement to License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety Analysis Including Adoption of Alternative Source Term," dated January 29, 2018 (ADAMS Accession No. ML18033B024).

9. McCoy, J. H., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Supplement to License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety Analysis Including Adoption of Alternative Source Term," dated April 19, 2018 (ADAMS Accession No. ML18114A115).

10. McCoy, J. H., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Supplement to License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety Analysis Including Adoption of Alternative Source Term," dated June 19, 2018 (ADAMS Accession No. ML18177A198).

11. McCoy, J. H., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Additional Supplement to License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety Analysis Including Adoption of Alternative Source Term," dated August 9, 2018 (ADAMS Accession No. ML18232A058).

12. McCoy, J. H., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Timeline for Response to RAI for License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety Analysis Including Adoption of Alternative Source Term," dated November 15, 2018 (ADAMS Accession No. ML18325A144).

13. McCoy, J. H., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Clean Revised Technical Specification Pages for License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety Analysis Including Adoption of Alternative Source Term," dated November 15, 2018 (ADAMS Accession No. ML18325A145).

14. Smith S. L., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Supplemental Response to RAI for License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety Analysis Including Adoption of Alternative Source Term," dated December 6, 2018 (ADAMS Accession No. ML18352A745).

15. Smith S. L., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Response to Request for Additional Information Related to Thermal Conductivity Degradation for License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety Analysis Including Adoption of Alternative Source Term," dated March 5, 2019 (ADAMS Accession No. ML19070A139).

16. Smith, S. L., Wolf Creek Nuclea rOperating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Implementation Date for License Amendment Request for the Transition to Westinghouse Core Design and Safety Analyses," dated May 2, 2019 (ADAMS Accession No. ML19133A057).

17. U. S. Nuclear Regulatory Commission, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Regulatory Guide 1.183, dated July 2000 (ADAMS Accession No. ML003716792).

18. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition":

    a. Chapter 2, Section 2.3.4, "Short-Term Atmospheric Dispersion Estimates for Accident Releases," Revision 3, dated March 2007 (ADAMS Accession No. ML070730398);

    b. Chapter 4, Section 4.2, "Fuel System Design," Revision 3, dated March 2007 (ADAMS Accession No. ML070740002);

    c. Chapter 6, Section 6.4, "Control Room Habitability System," Revision 3, dated March 2007 (ADAMS Accession No. ML070550069);

    d. Chapter 6, Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 4, dated March 2007 (ADAMS Accession No. ML070190178);

    e. Chapter 7, Section 7.0, "Instrumentation and Controls – Overview of Review Process," Revision 7, dated August 2016 (ADAMS Accession No. ML16020A049);

    f. Chapter 11, Section 11.3, "Gaseous Waste Management System," Revision 4, dated January 2016 (ADAMS Accession No. ML15029A039);

    g. Chapter 15, Section 15.0, "Introduction – Transient and Accident Analyses," Revision 3, dated March 2007 (ADAMS Accession No. ML070710376);

    h. Chapter 15, Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, dated July 2000 (ADAMS Accession No. ML003734190);

    i. Chapter 15, Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," Revision 2, dated July 1981 (ADAMS Accession No. ML052350147);

j. Chapter 16, Section 16.0, "Technical Specifications," Revision 3, dated March 2010 (ADAMS Accession No. ML100351425);

k. Chapter 18, Section 18.0, "Human Factors Engineering," Revision 3, dated December 2016 provides guidance to NRC reviewers regarding human factors issues, including manual operator actions. Section 18.0, Attachment A, "Guidance for Evaluating Credited Manual Operator Actions," specifies review guidance associated with evaluating the feasibility and reliability of operator actions (ADAMS Accession No. ML16125A114);

l. Branch Technical Position 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," Revision 4, dated January 2016. (ADAMS Accession No. ML15027A302).

19. U.S. Nuclear Regulatory Commission, "Standard Technical Specifications Westinghouse Plants," NUREG-1431, Revision 4, dated April 2102 (ADAMS Accession No. ML12100A222).

20. Wolf Creek Generating Station, Updated Final Safety Analysis Report, Revision 31 (ADAMS Package Accession No. ML18093A810).

21. Westinghouse Electric Corporation, WCAP-8339, "Westinghouse Emergency Core Cooling System Evaluation Model - Summary," dated June 1974 (ADAMS Accession No. ML033210318).

22. Westinghouse WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology," March 1978 (not-publicly availablable, proprietary information).

23. Not Used.

24. Pacific Northwest National Laboratory, "Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard," Report PNNL-18212, Revision 1, dated June 2011 (ADAMS Accession No. ML112070118).

25. U.S. Atomic Energy Commission, "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, dated March 23, 1962 (ADAMS Accession No. ML021720780).

26. U.S. Nuclear Regulatory Commission, "Meteorological Monitoring Programs for Nuclear Power Plants," Regulatory Guide 1.23, Revision 1, dated March 2007 (ADAMS Accession No. ML070350028).

27. U. S. Nuclear Regulatory Commission, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure," Regulatory Guide 1.24, Revision 0, dated March 1972 (ADAMS Accession No. ML083300020).

28. U. S. Nuclear Regulatory Commission, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.52, Revision 3, dated June 2001 (ADAMS Accession No. ML011710176).

29. U. S. Nuclear Regulatory Commission, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, Revision 1, dated November 1992 (ADAMS Accession No. ML003740205).

30. U. S. Nuclear Regulatory Commission, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Regulatory Guide 1.194, dated June 2003 (ADAMS Accession No. ML031530505).

31. U. S. Nuclear Regulatory Commission, "Control Room Habitability at Light-Water Nuclear Power Reactors," Regulatory Guide 1.196, Revision 1, dated January 2007 (ADAMS Accession No. ML063560144).

32. U. S. Nuclear Regulatory Commission, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Regulatory Guide 1.197, dated May 2003 (ADAMS Accession No. ML031490664).

33. U.S. Nuclear Regulatory Commission, "Functional Criteria for Emergency Response Facilities," Final Report, NUREG-0696, dated February 1981 (ADAMS Accession No. ML051390358).

34. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, dated November 1980 and "Clarification of TMI Action Plan Requirements - Requirements for Emergency Response Capability," NUREG-0737, Supplement No. 1, Reprinted February 1989 (ADAMS Accession Nos. ML102560051 and ML102560009, respectively).

35. U.S. Nuclear Regulatory Commission, "Experience with Implementation of Alternative Source Terms," Regulatory Issue Summary 2006-04, dated March 7, 2006 (ADAMS Accession No. ML053460347).

36. U. S. Nuclear Regulatory Commission, "Setpoints for Safety-Related Instrumentation," Regulatory Guide 1.105, Revision 3, dated December 1999 (ADAMS Accession No. ML993560062).

37. International Society of Automation, "Setpoints for Nuclear Safety Instrumentation," Part I of ISA-S67.04-1994.

38. U. S. Nuclear Regulatory Commission, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Regulatory Guide 1.109 Revision 1, dated October 1977 (ADAMS Accession No. ML003740384).

39. International Commission on Radiological Protection Publication 30, "Limits for Intakes of Radionuclides by Workers," 1979.[29]

40. Westinghouse Electric Company LLC, WCAP -17504, "Westinghouse Generic Setpoint Methodology," Revision 1, dated October 2016 (ADAMS Accession No. ML16314A091).

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[29] Copies of International Commission on Radiological Protection (ICRP) documents may be obtained through its Web site: http://www.icrp.org.

41. Broschak J. B., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: License Amendment Request for the Transition to Westinghouse Core Design and Safety Analysis," dated August 13, 2013 (ADAMS Accession No. ML13247A076).

42. U.S. Nuclear Regulatory Commission, "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, dated February 1995 (ADAMS Accession No. ML041040063).

43. Oak Ridge National Laboratory, "Iodine Evolution and pH Control," NUREG/CR-5950, dated December 1992 (ADAMS Accession No. ML063460464).

44. Pacific Northwest National Laboratory, "Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Revision 1, dated May 1997 (ADAMS Accession No. ML17213A187).

45. U.S. Nuclear Regulatory Commission, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data," NUREG-0917, dated July 1982 (ADAMS Accession No. ML12061A136).

46. Pacific Northwest Laboratory, "PAVAN: An Atmospheric-Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," NUREG/CR-2858, dated November 1982 (ADAMS Accession No. ML12045A149).

47. Singal, B., U.S. Nuclear Regulatory Commission, e-mail to Muilenburg, William T., WCNOC, "Request for Additional Information – License Amendment Request for Transition to Westinghouse Core Design and Safety Analyses including Adoption of Alternative Source Term (CAC No. MF9307)," dated June 14, 2017 (ADAMS Accession No. ML17166A038).

48. U.S. Environmental Protection Agency, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report No. 11, EPA-520/1-88-020, dated September 1988 (ADAMS Accession No. ML101590171).

49. U.S. Environmental Protection Agency, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report No. 12, EPA-402-R-93-081, dated September 1993 (ADAMS Accession No. ML101590169).

50. U.S. Environmental Protection Agency, "Exposure Factors Handbook: 2011 Edition," EPA/600/R-09/052F, dated September 2011 (ADAMS Accession No. ML14007A666).

51. Singal, B. K., U.S. Nuclear Regulatory Commission letter to Adam C. Heflin, Wolf Creek Nuclear Operating Corporation, "Wolf Creek Generating Station, Unit 1 – Regulatory Audit Summary Regarding License Amendment Request for Transition to Westinghouse Core Design and Safety Analyses including Adoption of Alternative Source Term (CAC No. MF9307; EPID L-2017-LLA-0211)," dated May 3, 2018 (ADAMS Accession No. ML18107A756).

52. U. S. Nuclear Regulatory Commission, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," Regulatory Guide 1.25, Revision 0, dated March 1972 (ADAMS Accession No. ML083300022).

53. Singal, B. K., U.S. Nuclear Regulatory Commission letter to Adam C. Heflin, Wolf Creek Nuclear Operating Corporation, "Wolf Creek Generating Station – Request for Additional Information RE: License Amendment Request for Transition to Westinghouse Core Design and Safety Analyses including Adoption of Alternative Source Term (CAC No. MF9307; EPID L-2017-LLA-0211)," dated December 4, 2017 (ADAMS Accession No. ML17331A178).

54. Westinghouse Electric Company LLC, Nuclear Safety Advisory Letter (NSAL)-07-11, "Decay Heat Assumption in Steam Generator Tube Rupture Margin-to-Overfill Analysis Methodology," dated November 15, 2007 (In Attachment 2 to the Donald C. Cook, Unit 1, Westinghouse Responses to NRC Request for Additional Information on Application for Amendment to Restore Normal Reactor Coolant System Pressure and Temperature Consistent with Previously Licensed Conditions) (ADAMS Accession No. ML14121A424).

55. Dietrich, A. W., U.S. Nuclear Regulatory Commission letter to Joel P. Gebbie, Indian Michigan Power Company, Nuclear Generation Group, "Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 – Revised Safety Evaluation for Amendment Nos. 332 and 314 RE: Adoption of TSTF-490, Rev. 0 and Implementation of Full-Scope Alternative Source Term (CAC Nos. MF8888 and MF8889)," dated December 28, 2016 (ADAMS Accession No. ML16327A110).

56. U. S. Nuclear Regulatory Commission, Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation," dated October 8, 2009 (ADAMS Accession No. ML0915505270).

57. Singal, B. K., U.S. Nuclear Regulatory Commission letter to Adam C. Heflin, Wolf Creek Nuclear Operating Corporation, "Wolf Creek Generating Station, Unit 1 – Request for Additional Information RE: License Amendment Request for Transition to Westinghouse Core Design Methodology for Selected Accident and Transient Analyses (CAC No. MF9307; EPID L-2017-LLA-0211)," dated October 4, 2018 (ADAMS Accession No. ML18270A094).

58. Morey, D. C., U.S. Nuclear Regulatory Commission, letter to Mr. James A. Gresham, Westinghouse Electric Company, "Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report WCAP-17642-P/NP, Revision 1, 'Westinghouse Performance Analysis and Design Model (PAD5),' (TAC No. MF3096)," dated September 28, 2017 (ADAMS Package Accession No. ML17338A396).

59. Westinghouse Electric Corporation, WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in UO_2 Fuel Rod," dated December 1989 (not publicly available, proprietary information). (Approved by the NRC staff by letter dated September 30, 1986.)

60. Westinghouse Electric Company LLC, WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," dated April 1999 (not publicly available, proprietary information).

61. Westinghouse Electric Corporation, WCAP-7907-P-A, "LOFTRAN Code Description," dated April 1984 (not publicly available, proprietary information). (Approved by the NRC staff by letter dated July 29, 1983).

62. Westinghouse Electric Corporation, WCAP-11397-P-A, "Revised Thermal Design Procedure," dated April 1989 (not publicly available, proprietary information).

63. Westinghouse Electric Company LLC, WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactors Non-LOCA Thermal-Hydraulic Safety Analysis," dated October 1999 (not-publicly available, proprietary information).

64. Westinghouse Electric Corporation, WCAP-7979-P-A, "TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code," dated January 1975 (not publicly available, proprietary information). (Approved by the NRC staff by letter dated July 29, 1974).

65. Westinghouse Electric Company Corporation, WCAP-10965-P-A, Revision 0, "ANC: A Westinghouse Advanced Nodal Computer Code," dated September 1986 (not publicly available; proprietary information).

66. Westinghouse Electric Corporation, WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," dated August 1987 (not publicly available, proprietary information).

67. Westinghouse Electric Company, WCAP-9226-P-A, Revision 1, "Reactor Core Response to Excessive Secondary Steam Releases," dated February 1988 (not publicly available, proprietary information).

68. Thadani, A., U.S. Nuclear Regulatory Commission, letter to W. J. Johnson, Westinghouse Electric Corporation, "Acceptance for Referencing of Licensing Topical Report WCAP-9226-P/9227-NP, 'Reactor Core Response to Excessive Secondary Steam Releases,'" dated January 31, 1989 (ADAMS Accession No. ML073521151).

69. Not Used.

70. Westinghouse WCAP-7588, Revision 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," dated January 1975.

71. Gramm, R. A., U.S. Nuclear Regulatory Commission, letter to James A. Gresham, Westinghouse Electric Company, "Suspension of NRC Approval for Use of Westinghouse Topical Report CENPD-254-P, "Post-LOCA Long-Term Cooling Model," Due to Discovery of Non-Conservative Modeling Assumptions During Calculations Audit," dated August 1, 2005 (ADAMS Accession No. ML051920310).

72. Combustion Engineering Inc., Topical Report CENPD-254-NP-A, "Post-LOCA Long Term Cooling Evaluation Model," dated June 1980 (not publicly available).

73. Collins, D. S., U.S. Nuclear Regulatory Commission, letter to James A. Gresham, Westinghouse Electric Company, "Clarification of NRC Letter Dated August 1, 2005, Suspension of NRC Approval for Use of Westinghouse Topical Report CENPD-254-P, 'Post-LOCA Long-Term Cooling Model,' Due to Discovery of Non-Conservative Modeling Assumptions During Calculations Audit," dated November 23, 2005 (ADAMS Accession No. ML052930272).

74. Pressurized Water Reactor Owner's Group, Presentation, "PWR Owners Group Boric Acid Precipitation Analysis Methodology Program," dated August 23, 2006 (ADAMS Accession No. ML062720565).

75. FirstEnergy Nuclear Operating Company, "Beaver Valley Power Station Extended Power Uprate Licensing Report," dated September 2004 (ADAMS Accession No. ML042920317).

76. Colburn, T. G., U.S. Nuclear Regulatory Commission, "Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) – Issuance of Amendment Regarding the 8-percent Extended Power Uprate (TAC Nos. MC4645 and MC4646)," dated July 19, 2006 (ADAMS Accession Package No. ML061720274).

77. Singal B. K., U.S. Nuclear Regulatory Commission, email to W.T. Muilenburg, Wolf Creek Nuclear Operating Corporation, "Request for Additional Information - License Amendment Request for Transition to Westinghouse Core Design and Safety Analyses (CAC No. MF9307)," September 21, 2017 (ADAMS Accession No. ML17265A014).

78. Kalynanam, N., U.S. Nuclear Regulatory Commission, letter to Joseph E. Venable, Entergy Operations, Inc., "Waterford Steam Electric Station, Unit 3 - Issuance of Amendment Re: Extended Power Uprate (TAC NO. MC1355)," April 15, 2005 (ADAMS Accession No. ML051030068).

79. Westinghouse Energy Systems Business Unit, Nuclear Safety Advisory Letter (NSAL)-95-001, "Minimum Cold Leg Recirculation Flow," dated January 12, 1995 (not publicly available, proprietary information).

80. Westinghouse, Nuclear Safety Advisory Letter (NSAL)-92-010, "Hot Leg Switchover Methodology."

81. Broschak, J. P., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Wolf Creek Nuclear Operating Corporation Proposed Path to Closure of Generic Safety Issue - 191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," dated May 16, 2013 (ADAMS Accession No. ML13148A272).

82. Westinghouse, Nuclear Safety Advisory Letter (NSAL)-00-016, "Rod Withdrawal from Subcritical Protection in Lower Modes."

83. Westinghouse Electric Company, WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," dated January 1999 (ADAMS Accession No. ML020430092).

84. U.S. Nuclear Regulatory Commission, NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988 (Legacy Library Accession No. 8802240276).

85. U. S. Nuclear Regulatory Commission, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," Regulatory Guide 1.195, dated May 2003 (ADAMS Accession No. ML031490640).

86. Smith S. L., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-482: Completion of Modification for License Amendment Request for the Transition to Westinghouse Core Design and Safety Analysis," dated May 15, 2019 (ADAMS Accession No. ML19148A571).

Attachment:
List of Acronyms

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Date: May 31, 2019

LIST OF ACRONYMS

AC	alternating current
ADAMS	Agencywide Documents Access and Management System
Am	Americium
ANS	American Nuclear Society
AOO	anticipated operational occurrence
ARC	alternative repair criteria
ARO	All-rod-out
ARV	atmospheric relief valves
AST	alternative source term
AV	allowable value
Ba	Barium
BAP	boric acid precipitation
BOP	balance-of-plant
Br	Bromine
Btu/lb_m	British thermal unit/pounds per mass
BWR	boiling-water reactor
°C	degree Celsius
cal/gm	calories per gram
CBE	control building envelope
CCP	centrifugal charging pump
Ce	Cerium
cfm	cubic feet per minute
CFR	*Code of Federal Regulations*
Cl	Chlorine
CLB	current licensing basis
Cm	Curium
COLR	Core Operating Limits Report
CRE	control room envelope
CREA	control rod ejection accident
CREVS	control room emergency ventilation system
CRVIS	control room ventilation isolation signal
Cs	Cesium
CsI	cesium iodide
CVCS	chemical and volume control system
DBA	design-basis accident
DCF	dose conversion factor
DE	Dose Equivalent
DF	decontamination factor
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
EAB	exclusion area boundary
ECCS	emergency core cooling system
EDE	effective dose equivalent
EES	emergency exhaust system
EOC	end of cycle
EPA	Environmental Protection Agency

ESF	engineered safety feature
ESFAS	engineered safety feature actuation system
Eu	Europium
°F	degree Fahrenheit
FGR	Federal Guidance Report
FHA	fuel handling accident
Ft^3	cubic feet
g-moles	gram moles
GDC	General Design Criterion/Criteria
gm/cc	gram per cubic centimeter
gpd	gallons per day
gpm	gallons per minute
GSI	Generic Safety Issue
GWD/MTU	gigawatt days per metric ton of uranium
HCl	hydrochloric acid
He	Helium
HFP	hot full power
HVAC	heating, ventilation, and air conditioning
HZP	hot zero power
H_3BO_3	boric acid
HNO_3	nitric acid
I-131	Iodine-131
ICRP	International Commission on Radiological Protection
IEEE	Institute of Electrical and Electronics Engineers
IN	Information Notice
IHSI	intermediate head safety injection
ISA	International Society of Automation
JFD	joint frequency distribution
K_{eff}	reactivity coefficient
Kr	Krypton
kw/ft	kilowatt per foot
La	Lanthanum
LAR	license amendment request
LBLOCA	large-break loss-of-coolant accident
lb_m	pounds mass
lb_m/hr	pound mass per hour
lb_m/ft^3	pounds mass per cubic foot
LCO	limiting condition for operation
LHGR	linear heat generation rate
LLB	letdown line break
LOAC	loss of non-emergency AC
LOCA	loss-of-coolant accident
LOL/TT	loss of load/turbine trip
LOOP	loss of offsite poser
LPZ	low population zone
LRA	locked rotor accident
LTCC	long-term core cooling
LWR	light-water reactor
LWT	liquid waste tank

m	meter
MMF	MINIMUM MEASURED FLOW
Mo	Molybdenum
mrem	millirem
Mrad	Megarad
MSLB	main steam line break
MSSV	main steam safety valve
MTU	metric ton of uranium
MUR	measurement uncertainty recapture
MWD/MTU	megawatt days per metric ton of uranium
MWt	megawatt thermal
NaOH	sodium hydroxide
Nb	Niobium
Nd	Neodymium
Np	Neptunium
NRC	U.S. Nuclear Regulatory Commission
NSAL	Nuclear Safety Advisory Letter
NSHC	no significant hazards consideration
Pd	Palladium
pH	scale of acidity
Pm	Promethium
PNNL	Pacific Northwest National Laboratory
ppm	parts per million
Pr	Praseodymium
psia	pounds per square inch absolute
psig	pounds per square inch gauge
Pu	Plutonium
PWR	pressurized-water reactor
RAI	request for additional information
RCCA	rod cluster control assembly
RCP	reactor coolant pump
RCS	reactor coolant system
rem	roentgen equivalent man
RG	Regulatory Guide
Rh	Rhodium
RHR	residual heat removal
RIS	Regulatory Issue Summary
Rn	Radon
RTDP	revised thermal design procedure
RTP	rated thermal power
RTS	reactor trip system
Ru	Ruthenium
RWST	refueling water storage tank
SAT	spray additive tank
Sb	Antimony
SBLOCA	small-break loss of coolant accident
scfm	standard cubic feet per minute
Se	Selenium
SE	safety evaluation

SG	steam generator
SGTR	steam generator tube rupture
SI	safety injection
SL	safety limit
SLB	steam line break
Sm	Samarium
SR	surveillance requirement
SRP	Standard Review Plan
STS	Standard Technical Specification
Sv	sievert
Tc	Technetium
TCD	thermal conductivity degradation
Te	Tellurium
TEDE	total effective dose equivalent
T/H	thermal/hydraulics
TID	Technical Information Document
TMI	Three Mile Island
TR	topical report
TS	Technical Specification
TSC	technical support center
TSTF	Technical Specifications Task Force
U-235	Uranium-235
UFSAR	Updated Final Safety Analysis Report
UO_2	uranium dioxide
VCT	volume control tank
VFTP	ventilation filter testing program
WCGS	Wolf Creek Generating Station, Unit 1
WCNOC	Wolf Creek Nuclear Operating Corporation
WGDT	waste gas decay tank
WLOP	Westinghouse low pressure
XE-133	Xenon-133
Zr	Zirconium

SUBJECT: WOLF CREEK GENERATING STATION, UNIT 1 - ISSUANCE OF AMENDMENT NO. 221 RE: TRANSITION TO WESTINGHOUSE CORE DESIGN AND SAFETY ANALYSES INCLUDING ADOPTION OF ALTERNATIVE SOURCE TERM (CAC NO. MF9307; EPID L-2017-LLA-0211) DATED MAY 31, 2019

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ADAMS Accession No. ML19100A122 ***via memo dated **via e-mail dated**

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DATE	2/26/18	1/31/19	8/30/18	12/11/17	11/27/17
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