WOLF CREEK
NUCLEAR OPERATING CORPORATION

Jaime H. McCoy
Site Vice President

November 15, 2018

WO 18-0044

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

Reference:
1) Letter ET 17-0001, dated January 17, 2017, from J. H. McCoy, WCNOC, to USNRC
2) Letter ET 17-0011, dated May 4, 2017, from J. H. McCoy, WCNOC, to USNRC
3) Letter WO 18-0004, dated January 15, 2018, from C. O. Reasoner, WCNOC, to USNRC
4) Letter ET 18-0018, dated June 19, 2018, from J. H. McCoy, WCNOC, to USNRC

Subject: 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

To Whom It May Concern:

Reference 1 provided the Wolf Creek Nuclear Operating Corporation (WCNOC) application to revise the Wolf Creek Generating Station (WCGS) Technical Specifications (TS). The proposed amendment would support transition to the Westinghouse Core Design and Safety Analysis methodologies. In addition, the license amendment request (LAR) included revising the WCGS licensing basis by adopting the Alternative Source Term radiological analysis methodology in accordance with 10 CFR 50.67, "Accident Source Term." As part of the original LAR, a number of revised TS pages were submitted. Since that time, the Nuclear Regulatory Commission (NRC) staff has provided a number of requests for additional information (RAIs) related to this LAR. References 2, 3, and 4 provided WCNOC's responses to RAIs which included revised TS pages. In some cases, these revised TS pages were revising TS pages that were previously submitted. The intent of this submittal is to provide all of the clean, revised TS pages associated with this LAR in their most current form. The enclosure to this submittal provides these TS pages.

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NRR

The information provided in this submittal does not expand the scope of the application and does not impact the no significant hazards consideration determination presented in Reference 1.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," a copy of this submittal is being provided to the designated Kansas State official.

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4156, or Cynthia R. Hafenstine at (620) 364-4204.

Sincerely,

Jaime H. McCoy

JHM/rlt

Enclosure: Wolf Creek Westinghouse Methodology and AST LAR Revised Technical Specification Pages

cc: K. M. Kennedy (NRC), w/e
B. K. Singal (NRC), w/e
K. S. Steves (KDHE), w/e
N. H. Taylor (NRC), w/e
Senior Resident Inspector (NRC), w/e

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Jaime H. McCoy, of lawful age, being first duly sworn upon oath says that he is Site Vice President of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Jaime H. McCoy
Jaime H. McCoy
Site Vice President

SUBSCRIBED and sworn to before me this 15th day of November, 2018.

GAYLE SHEPHEARD
My Appointment Expires
July 24, 2019

Gayle Shepheard
Notary Public

Expiration Date 7/24/2019

ENCLOSURE TO
WO 18-0044

Wolf Creek Westinghouse Methodology and AST LAR
Revised Technical Specification Pages
(34 pages)

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 $\geq$ 1.17 for the WRB-2 DNB correlation, and $\geq$1.13 for the ABB-NV DNB correlation, and $\geq$ 1.18 for the WLOP DNB.

2.1.1.2 The peak centerline temperature shall be maintained $\leq$ 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 $\leq$ 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
	<u>OR</u>		
	A.2	Initiate action to place the Rod Control System in a condition incapable of rod withdrawal.	Immediately
	<u>OR</u>		
	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.	72 hours
	OR	
	P.2 Reduce THERMAL POWER to < P-9.	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.	24 hours
	OR	
	Q.2 Be in MODE 3.	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.	24 hour
	OR	
	R.2 Be in MODE 3.	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
	<u>OR</u>		
	V.2.1	B in MODE 2 with k_{eff} < 1.0.	78 hours
	<u>AND</u>		
	V.2.2.1	Initiate action to fully insert all rods.	78 hours
		<u>AND</u>	
	V.2.2.2	Initiate action to place the Rod Control System in a condition incapable of rod withdrawal.	78 hours
	<u>OR</u>		
	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 incore detector 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 incore detector 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.
(e) Below 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 termperatures ≥ 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 $\leq$ 60 μCi/gm.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT XE-133 not within limit <u>OR</u> DOSE EQUIVALENT I-131 > 60 μCi/gm.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	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 <u>AND</u> 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
	<u>AND</u>	
	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
	<u>AND</u>	
	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. AND C.2 Be in MODE 5.	6 hours 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. OR D.2.1 Suspend CORE ALTERATIONS. AND D.2.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately Immediately
E. Two CREVS trains inoperable during movement of irradiated fuel assemblies or during CORE ALTERATIONS. OR 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.	E.1 Suspend CORE ALTERATIONS. AND E.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately

(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.	Immediately
	<u>AND</u>		

(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.
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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 Programs and Manuals

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-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)