WCNOC-NRC Pre-application Meeting

Core Design and Safety Analysis Methodology Transition License Amendment Request Revised

August 25, 2016

Wolf Creek Nuclear Operating Corporation

Meeting Agenda

- Meeting Purpose / Objectives
- Introductions
- History of Original LAR
- Revised LAR content
  - Transition to Westinghouse Analysis Methodologies
  - Alternative Source Term (AST)
- LAR Schedule – WCAP-17504 approval for use
- Summary

Introductions

- NRC
- WCNOC / Westinghouse Team

History of Original LAR

- Original LAR submitted August 13, 2013
- LAR withdrawn June 18, 2014
- NRC issued RAIs on December 13, 2013, March 5, 2014, April 3, 2014 and April 30, 2014
  - RAIs issued on December 13 and March 5 were previously responded to
  - A response to all applicable RAIs has been included in the revised LAR and integrated into the text, where appropriate

Revised LAR Content

- Revised LAR based on:
  - Transition to Westinghouse core design and safety analysis methodologies
  - Full Scope Implementation of Alternative Source Term (AST)
  - AST uses input data from the safety analysis transients and thus they are being submitted together
- Revised LAR no longer based on:
  - Transition to Westinghouse Setpoint Methodology (WSM) for RTS/ESFAS
- WSM will be used for Revised Thermal Design Procedure uncertainties

Revised LAR Content

- LAR remains consistent with NEI 06-02, "License Amendment Request (LAR) Guidelines" template
- Attachments:
  - I - Evaluation
  - II - Proposed TS Changes (Mark-up)
  - III - Revised TS pages
  - IV - Proposed TS Bases Changes (info only)
  - V - Proposed COLR Changes (info only)

Revised LAR Content (cont.)

- Enclosures:
  - I - WCAP-17658-NP – Transition Licensing Report
  - II - WCAP-18083-P – RTDP Uncertainty Calcs for WCGS
  - III - WCAP-18083-NP
  - IV - Full Scope Implementation of AST
  - V - CD-ROM containing Meteorological Data
  - VI - NRC RAI Responses - Proprietary
  - VII - NRC RAI Responses – Non-Proprietary
  - VIII - Supplemental Info to Support NRC RAI Responses
  - IX - Proprietary Information Affidavit for WCAP-18083-P
  - X - Proprietary Information Affidavit for Enclosure VI

Transition to Westinghouse Analysis Methodologies

- Core Design and Safety Analysis Methodology
  - The Non-LOCA safety analyses were analyzed with Westinghouse, NRC approved methods
  - All of the Westinghouse Non-LOCA methods are applicable to the WCGS
  - The Containment response analyses of record are not impacted
  - The SBLOCA and LBLOCA analyses of record are not impacted
  - The Core Design and Fuel Rod Design will be evaluated for each reload cycle

Transition to Westinghouse Analysis Methodologies

- Overview of Analysis Methodology Scope
  - Non-LOCA Safety Analyses
    - ✓ RETRAN-02 for Westinghouse PWRs (WCAP-14882-P-A) was used for majority of the analyses
    - ✓ LOFTRAN (WCAP-7907-P-A) was used for some analyses
    - ✓ Other Codes that were used:
      - TWINKLE (WCAP-7979-P-A)
      - FACTRAN (WCAP-7908-A)
  - Non-LOCA Thermal-Hydraulics (T&H) Safety Analyses
    - ✓ VIPRE-01 (WCAP-14565-P-A) was used for the T&H analyses
  - RETRAN-02 (WCAP-14882-P-A) and WCAP-10698-P-A were used for the Steam Generator Tube Rupture Margin to Overfill and Input to the Dose analyses

Transition to Westinghouse Analysis Methodologies

- Overview of Analysis Methodology Scope (cont.)
  - DNB Correlations used in the VIPRE-01 DNBR calculations
    - ✓The WRB-2 DNB correlation will continue to be used as the primary DNB correlation for the T&H analyses of fuel regions above the first mixing vane grid
    - ✓The ABB-NV DNB correlation was used for the T&H analyses of fuel regions below the first mixing vane grid
    - ✓The WLOP DNB correlation was used for the T&H analyses that are outside the range of applicability of the WRB-2 and ABB-NV DNB correlations

Transition to Westinghouse Analysis Methodologies

- Implementation of the Westinghouse Non-LOCA Safety Analysis Methodology resulted in six changes to the WCGS TSs
  - SLs 2.1.1, Added the ABB-NV and WLOP DNB Correlations
  - TS 3.1.9, RCS Boron Concentration – New LCO
  - TS 3.3.1, RTS Function 10, Reactor Coolant Flow- Low
  - TS 3.4.1, RCS Pressure, Temperature and Flow DNB Limits

Transition to Westinghouse Analysis Methodologies

- Implementation of the Westinghouse Non-LOCA Safety Analysis Methodology results in six changes to the WCGS TSs (cont.)
  - TS Table 3.7.1-1, OPERABLE MSSVs versus Maximum Allowable Power, the maximum allowable power for 4, 3, and 2 OPERABLE MSSVs was revised
  - TS 5.6.5, COLR
    - ✓ Added WCAP-9272-P-A, the Westinghouse Reload Methodology to Specification 5.6.5 b
    - ✓ Deleted the WCNOC methodologies from Specification 5.6.5 b
    - ✓ WCAP-9272-P-A is the only methodology associated with a Tech Spec COLR parameter

Transition to Westinghouse Analysis Methodologies

- New TS 3.1.9 was added to address Westinghouse NSAL-00-016 which was covered as a part of the transition
- TS changes are in accordance with TSTF-453-T, Rev. 2, "Addition of New Tech Spec on RCS Boron Limits and Revisions to Tech Spec 3.3.1 to address RWFS"

Transition to Westinghouse Analysis Methodologies

- Transition includes Revised Thermal Design Procedure (RTDP) instrument uncertainty previously controlled by WCNOC
- Enclosure II - WCAP-18083-P – RTDP Uncertainty Calcs for WCGS
- Previously, these RTDP parameter uncertainties were submitted and reviewed by NRC as part of the instrumentation setpoint methodology transition
- WCAP-18083-P relies on NRC approval of WCAP-17504-P, "WESTINGHOUSE GENERIC SETPOINT METHODOLOGY"
- Our understanding is that there are Conditions in the Draft Safety Evaluation for WCAP-17504-P, and these will be generically addressed within the LAR

Instrumentation Setpoint Uncertainty Analysis

- Change requests associated with RTS/ESFAS TSTF-493, Option A, have been removed from the revised LAR

Alternative Source Term

- Full Scope Implementation of the AST
  - Radiological dose consequences analyses performed for the accidents specified in Regulatory Guide (RG) 1.183 include:
    - ✓ Main Steamline Break (USAR Section 15.1.5.3)
    - ✓ Locked Rotor (USAR Section 15.3.3.3)
    - ✓ Rod Ejection (USAR Section 15.4.8.3)
    - ✓ Steam Generator Tube Rupture (USAR Section 15.6.3.3)
    - ✓ Loss of Coolant Accident (USAR Section 15.6.5.4)
    - ✓ Fuel Handling Accident (USAR Section 15.7.4)

Alternative Source Term

- Full Scope Implementation of the AST (cont.)
  - Radiological dose consequences analyses performed for additional accidents not specified in RG 1.183 include:
    - ✓Loss of Non-Emergency AC Power (USAR Section 15.2.6.3)
    - ✓Letdown Line Break (USAR Section 15.6.2.1)
    - ✓Waste Gas Decay Tank Failure (USAR Section 15.7.1)
    - ✓Liquid Waste Tank Failure (USAR Section 15.7.2)
  - Dose consequences analyses were performed using version 3.03 of the RADTRAD computer code

Alternative Source Term

- Full Scope Implementation of AST (cont.)
  - No changes to the licensing basis EQ dose analyses – maintaining the TID-14844 accident source term
  - No changes to the licensing basis NUREG-0737 evaluations other than the Control Room Habitability Envelope (CRHE) doses (III.D.3.4) and Technical Support Center doses (III.A.1.2)

Alternative Source Term

- Atmospheric Dispersion Factors (χ/Q)
  - χ/Q values were calculated
    - ✓ Offsite (EAB and LPZ) χ/Q values were calculated using the PAVAN code consistent with RG 1.145
    - ✓ Control Room and TSC χ/Q values were calculated using the ARCON96 code consistent with RG 1.194
  - Meteorological Data
    - ✓ Five years of WCGS site-specific meteorological data was collected
    - ✓ Data recovery for the 5-year period met the 90% recovery criterion of RG 1.23

Alternative Source Term

- Current licensing basis changes
  - Revises USAR Chapter 15 dose analysis for 10 accidents (includes the 6 DBAs in RG 1.183)
  - Offsite, Control Room, and TSC atmospheric dispersion factors based on site-specific meteorological data
  - Revises the CRHE unfiltered in-leakage from 20 scfm to 50 scfm
  - Revises the Control Building unfiltered in-leakage from 300 scfm to 400 scfm
  - TS changes to address the update of the accident source term and associated DBAs
  - TS changes to address the adoption of TSTF-51-A, Revision 2 have been removed from the revised LAR
  - A modification to the control room HVAC system isolation dampers will be implemented to eliminate a local TCA

Alternative Source Term

- Technical Specification Changes
  - Definition of DOSE EQUIVALENT I-131
    - ✓ Revised to only allow the use of the dose conversion factors from EPA Federal Guidance Report No. 11
  - Definition of DOSE EQUIVALENT XE-133
    - ✓ Revised to only allow the use of the dose conversion factors EPA Federal Guidance Report No. 12
  - TS SR 3.3.7.6, CREVS Instrumentation – Hi rad channel response time testing – New
    - ✓ Identified while responding to RAIs and reviewing other AST submittals
  - Specification 5.5.12, "Gas and Storage Tank Radioactivity Monitoring Program"
    - ✓ Revises the limit of radioactivity contained in each gas storage tank to be less than the revised dose limit of 0.1 rem (current limit is 0.5 rem)
  - Specification 5.5.18, "Control Room Habitability Program"
    - ✓ Revised to recognize TEDE terminology

Alternative Source Term

- Enclosure IV of LAR – Sections

1 - DESCRIPTION
2 - PROPOSED CHANGES
3 - BACKGROUND
4 - TECHNICAL ANALYSIS
5 - RG 1.183 CONFORMANCE TABLE
6 - RG 1.145 CONFORMANCE TABLE
7 - RG 1.194 CONFORMANCE TABLE
8 - RIS 2006-04 TABLE
9 - PROPOSED TS MARKUPS
10 - RETYPED TS PAGES
11 - PROPOSED BASES MARKUPS (information only)
12 - PROPOSED TRM and BASES MARKUP (information only)
13 - PROPOSED USAR CHANGES (information only)

Schedule

- Submit LAR to NRC — 4th Qtr 2016
- Requested Approval Date — February 23, 2018
- Start of Refueling Outage 22 — March 17, 2018

Summary

- WCNOC intends to submit a LAR in 4th qtr 2016 to revise the WCGS TSs based on:
  - Transition to the Westinghouse core design and safety analysis methodologies
  - Full Scope Implementation of the Alternative Source Term (AST)
- Schedule dependent upon NRC approval and issuance of acceptable for use letter for WCAP-17504
- Scope of LAR reduced; largely repetitive to what has been previously reviewed; RAIs addressed
- Request NRC approval prior to February 23, 2018 to allow for implementation during RFO 22

Closing

- Questions?

BACKUP SLIDES

Transition to Westinghouse Analysis Methodologies

- Safety Limits (SLs) changes

2.0 SAFETY LIMITS (SLs)

2.1 SLs

≥ 1.17

≥ 1.13 for the ABB-NV DNB correlation, and ≥ 1.18 for the WLOP DNB

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 ≥ ~~1.23~~ for the WRB-2 DNB correlation, and ≥ ~~1.30 for the W-3 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.

Transition to Westinghouse Analysis Methodologies

- New Reactivity Control Systems TS 3.1.9

(continued)

Insert New LCO 3.1.9

Insert New Page

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.

Transition to Westinghouse Analysis Methodologies

- New Reactivity Control Systems TS 3.1.9

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

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Transition to Westinghouse Analysis Methodologies

- New Reactivity Control Systems TS 3.1.9

Insert New Page

RCS Boron Limitations < 500°F
3.1.9

SURVEILLANCE REQUIREMENTS

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

Transition to Westinghouse Analysis Methodologies

- Insert New TS 3.3-9

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

INSERT 3.3-9

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE	FREQUENCY

Transition to Westinghouse Analysis Methodologies

- New 3.3-9

INSERT 3.3-9

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 Be 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

Transition to Westinghouse Analysis Methodologies

- New 3.3-9 (Continued)

INSERT 3.3.9 (continued)

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

Transition to Westinghouse Analysis Methodologies

- Insert 3.3-15A and B

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
INSERT 3.3-15A					
~~b. Low~~	~~1[c],2~~	~~4~~	~~E~~	~~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
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

(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.

INSERT 3.3-15B

Transition to Westinghouse Analysis Methodologies

- Insert to TS 3.3.1

INSERT 3.3-15A

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE [a]
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

INSERT 3.3-15B

(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 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.

Transition to Westinghouse Analysis Methodologies

- TS Table 3.3.1-1, Function 10

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%~~[m]~~ of Normalized Flow
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

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.
(g) Above the P-7 (Low Power Reactor Trips Block) interlock.
~~(m) % of design flow - 90,324 gpm.~~

Transition to Westinghouse Analysis Methodologies

- TS 3.4.1 changes

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.

361,200

APPLICABILITY: MODE 1.

Transition to Westinghouse Analysis Methodologies

- TS 3.4.1 changes (Continued)

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 ≥ ~~37.1 x 10^4~~ 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 ≥ ~~37.1 x 10^4~~ 361,200 gpm and greater than or equal to the limit specified in the COLR.	18 months

Transition to Westinghouse Analysis Methodologies

- TS Table 3.7.1-1

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	~~87~~ 70
3	~~65~~ 51
2	~~44~~ 31

Transition to Westinghouse Analysis Methodologies

- Specification 5.6.5 b.

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. ~~WCNOC Topical Report TR 90-0025 W01, "Core Thermal Hydraulic Analysis Methodology for the Wolf Creek Generating Station."~~

1

2. WCAP-11397-P-A, "Revised Thermal Design Procedure."

3. ~~WCNOC Topical Report NSAG-006, "Transient Analysis Methodology for the Wolf Creek Generating Station."~~

(continued)

Wolf Creek - Unit 1 5.0-25 Amendment No. ~~123, 142, 144, 158, 159, 164~~, 179

Transition to Westinghouse Analysis Methodologies

- Specification 5.6.5 b. (cont.)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

~~4.~~ WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification."

~~5.~~ ~~WCNOC Topical Report NSAG-007, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station."~~

WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology."

~~6.~~ ~~NRC Safety Evaluation Report dated March 30, 1993, for the "Revision to Technical Specification for Cycle 7."~~

~~7.~~ WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)."

~~8.~~ WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON."

~~9.~~ WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology."

~~10.~~ WCAP 10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code."

~~11.~~ WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report."

~~12.~~ WCAP-8745-P-A, "Design Bases for the Thermal Power ΔT and Thermal Overtemperature ΔT Trip Functions."

Transition to Westinghouse Analysis Methodologies

- COLR Change

Attachment V of ET 16-00XX
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WOLF CREEK NUCLEAR OPERATING CORPORATION

Wolf Creek Generating Station
Cycle ~~21~~ 2X
Core Operating Limits Report
Revision 1

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

Parameter	Indicated Value
Pressurizer pressure	Pressure ≥ ~~2220~~ psig
RCS average temperature	T_{avg} ≤ ~~590.5 °F~~
RCS total flow rate	Flow ≥ ~~371,000~~ gpm

2219 (Average of 4 channels)
2221 (Average of 3 channels)

590.8°F (Average of 4 channels)
590.6°F (Average of 3 channels)

376,000

2.11 Boron Concentration (LCO 3.9.1)

The refueling boron concentration shall be greater than or equal to 2300 ~~PPM.~~ ppm

2.12 SHUTDOWN MARGIN (LCO 3.1.1, 3.1.4, 3.1.5, 3.1.6, & 3.1.8)

The SHUTDOWN MARGIN shall be greater than or equal to 1300 pcm (1.3% Δk/k).

~~2.13 Departure from Nucleate Boiling Ratio (DNBR) Limits (B 3.4.1, ASA)~~

~~Safety Analysis DNBR Limit~~	~~1.76~~
~~WRB-2 Design Limit DNBR~~	~~1.23~~

Delete Section 2.13

Alternative Source Term

- Definition of DOSE EQUIVALENT I-131

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~~:~~ ~~1) Table III of TID-14844, AEC, 1962, Calculation of Distance Factors for Power and Test Reactor Sites," or~~ ~~2) Table E-7 of Regulatory Guide 1.109, Revision 1, NRC, 1977, or~~

1.1 Definitions (continued)

DOSE EQUIVALENT I-131 (continued)	~~3) ICRP 30, 1979, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity," or~~ ~~4)~~ 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."

Alternative Source Term

- Definition DOSE EQUIVALENT XE-133

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," ~~or using the dose conversion factors from Table B-1 of Regulatory Guide 1.109, Revision 1, NRC, 1977~~.

Alternative Source Term

- SR 3.3.7.6

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

Alternative Source Term

- SR 3.3.7.6 (Continued)

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, and (a)	2	SR 3.3.7.4	NA
2.	Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	1, 2, 3, 4, and (a)	2 trains	SR 3.3.7.3	NA
3.	Control Room Radiation-Control Room Air Intakes	1, 2, 3, 4, and (a)	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.5	(b)
4.	Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a, for all initiation functions and requirements.			

SR 3.3.7.6

Alternative Source Term

- TS 5.5.12

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 ~~a whole body exposure of~~ ≥ ~~0.5~~ 0.1 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

TEDE

Alternative Source Term

- TS 5.5.18

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 ~~whole body or its equivalent to any part of the body~~ for the duration of the accident. The program shall include the following elements:

TEDE

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

List of Acronyms

AC	Alternating Current
AST	Alternative Source Term
COLR	Core Operating Limits Report
CREVS	Control Room Emergency Ventilation System
CRHE	Control Room Habitability Envelope
DBA	Design Basis Accident
DNB	Departure from Nucleate Boiling
EAB	Exclusion Area Boundary
EPA	Environmental Protection Agency
EQ	Environmental Qualification
LAR	License Amendment Request
LBLOCA	Large Break LOCA
LCO	Limiting Condition of Operation
LPZ	Low Population Zone
MSSV	Main Steam Safety Valve
NEI	Nuclear Energy Institute
Non-LOCA	Non- Loss of Coolant Accident
NRC	Nuclear Regulatory Commission
NSAL	Nuclear Safety Advisory Letter
NSSS	Nuclear Steam Supply System
RAI	Request for Additional Information
RG	Regulatory Guide
RCS	Reactor Coolant System
RFO	Refueling Outage
RTDP	Revised Thermal Design Procedure
RTS	Reactor Trip System
RTS/ESFAS	Reactor Trip System/Engineered Safety Features Actuation System
RWFS	Rod Withdrawal from Subcritical
SBLOCA	Small Break LOCA
SL	Safety Limits
SR	Surveillance Requirement
TCA	Time Critical Action
TEDE	Total Effective Dose Equivalent
TID	Technical Information Document
T&H	Thermal – Hydraulic
TS	Technical Specifications
TSC	Technical Support Center
TSTF	Technical Specification Task Force
USAR	Updated Safety Analysis Report
WCGS	Wolf Creek Generating Station
WCNOC	Wolf Creek Nuclear Operating Corporation
WSM	Westinghouse Setpoint Methodology
X	Chi