EVALUATION

EVALUATION

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Wolf Creek Nuclear Operating Corporation (WCNOC) hereby requests an amendment to Renewed Facility Operating License No. NPF-42 for the Wolf Creek Generating Station (WCGS). The proposed license amendment request (LAR) would implement the following changes:

1. Replace the existing WCNOC methodology (developed by WCNOC) for performing core design, non-loss-of-coolant-accident (non-LOCA) and LOCA safety analyses (for Post-LOCA Subcriticality and Cooling only) to the standard Westinghouse Nuclear Regulatory Commission (NRC) approved methodologies for performing these analyses. This proposed change would result in revisions to the following Specifications:

    - 2.1.1, "Reactor Core SLs,"
    - 3.1.9, "RCS Boron Limitations < 500°F,"
    - 3.3.1, "Reactor Trip System (RTS) Instrumentation,"
    - 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits,"
    - 3.7.1, "Main Steam Safety Valves (MSSVs)," and
    - 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)."

2. This amendment request also proposes to revise the WCGS licensing basis by adopting the Alternative Source Term (AST) radiological analysis methodology in accordance with 10 CFR 50.67, "Accident source term." This amendment request is for a full scope implementation of the AST as described in NRC Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0. Enclosure IV of this amendment request contains a complete description of the proposed AST changes including supporting information, the TS and TS Bases markups, and evaluations of the changes. The AST portion of this amendment request was included in a separate enclosure (Enclosure IV) to facilitate a separate review of the AST changes, independent from the other proposed changes in this amendment request.

3. As listed in item 1 above, the proposed amendment would add a new Technical Specification (TS) Limiting Condition for Operation (LCO) 3.1.9, "RCS Boron Limitations <500°F." This amendment assures that the required mitigative capability is available, in the form of adequate SHUTDOWN MARGIN (SDM) or an automatic reactor trip, for an uncontrolled rod cluster control assembly (RCCA) bank withdrawal event that may be postulated to occur during low power or subcritical (startup) conditions.

2.0 DETAILED DESCRIPTION

WCNOC previously submitted a LAR on August 13, 2013 (Reference 8) requesting approval of changes to the WCGS TSs. That application proposed the transition to Westinghouse core design and safety analysis methodologies, full scope implementation of AST, and implementation of instrumentation setpoint and control uncertainty calculations based on the current Westinghouse Setpoint Methodology (including adoption of Option A of Technical Specification Task Force (TSTF) TSTF-493-A, Revision 4). For the current application, WCNOC is not requesting approval of changes to the instrumentation setpoint and control uncertainty methodology.

WCNOC subsequently withdrew the LAR on June 18, 2014 (Reference 9) based on deficiencies discovered by WCNOC. WCNOC responses to Requests for Additional Information (RAI) submitted by References 10, 11, and 12 have been incorporated into the current application. There are two RAIs (References 13 and 14) that WCNOC did not respond to due to the withdrawal of Reference 8. Enclosures VI and VII provide the WCNOC responses to the questions in the two RAIs and the applicable information has been incorporated into the current application. Enclosure VIII provides supplemental documents that were requested by the RAIs.

CORE DESIGN AND SAFETY ANALYSIS METHODOLOGY TRANSITION

The transition from the existing WCNOC methodology (developed by WCNOC) for performing core design, non-LOCA and LOCA safety analyses (for Post-LOCA Subcriticality and Cooling only) to the standard Westinghouse NRC approved methodologies for performing these analyses is described in detail in Enclosure I of this LAR. Enclosure I of this LAR contains WCAP-17658-NP, Revision 1, "Wolf Creek Generating Station Transition of Methods for Core Design and Safety Analyses – Licensing Report."

The TS changes resulting from the transition to Westinghouse methodologies are described below.

1. Safety Limits (SLs) 2.1.1 "Reactor Core SLs," states:

   > "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:

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

   The Allowable Value for TS Table 3.3.1-1 Function 10, "Reactor Coolant Flow - Low" is:

≥ 88.9% of design flow - 90,324 gpm

The proposed change would revise the TS Allowable Value to:

"≥ 88.9% of Normalized Flow"

The proposed change to TS Table 3.3.1-1 Function 10, would delete Footnote "m" which identifies the "% of design flow - 90,324 gpm" portion of the Allowable Value.

3. TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," Limiting Condition for Operation (LCO) states:

"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 ≥ 37.1×10^4 gpm and greater than or equal to the limit specified in the COLR. "

The proposed change would revise the minimum measured Reactor Coolant System (RCS) flow specified in TS 3.4.1 LCO Part c. from 37.1×10^4 gpm to 376,000 gpm. The new value for minimum measured flow (376,000 gpm) would then be relocated to the CORE OPERATING LIMITS REPORT (COLR). The RCS flow value specified in TS 3.4.1 LCO Part c. would be replaced by the RCS thermal design flow (TDF) of 361,200 gpm. The RCS TDF flow value of 361,200 gpm would also replace the current RCS flows specified in Surveillance Requirement (SR) 3.4.1.3 and SR 3.4.1.4 of TS 3.4.1.

4. TS 3.7.1, "Main Steam Safety Valves (MSSVs)," LCO requires 5 OPERABLE MSSVs per steam generator (SG). TS 3.7.1 Table 3.7.1-1, "OPERABLE Main Steam Safety Valves versus Maximum Allowable Power," specifies the power limits (in % RATED THERMAL POWER (RTP)) applicable when the number of OPERABLE MSSVs per SG is less than 5. Table 3.7.1-1 specifies the following limits:

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	**MAXIMUM ALLOWABLE POWER (% RTP)**
4	87
3	65
2	44

The proposed change would revise Table 3.7.1-1 as follows:

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

5. Specification 5.6.5, “CORE OPERATING LIMITS REPORT (COLR),” Section b. lists the analytical methods used to determine the core operating limits.

   The proposed change would delete the following WCNOC related analytical methods listed in Section b. of Specification 5.6.5:

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

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

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

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

   The proposed change would add the following Westinghouse analytical method to those listed in Section b. of Specification 5.6.5:

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

   Due to the changes described above, the list of analytical methods in Specification 5.6.5 will be renumbered as applicable.

   WCAP-9272 (Reference 1), the Westinghouse Reload Methodology, which is being added to TS 5.6.5, is the only methodology that is associated with the determination of a TS COLR parameter.

   The other NRC approved methodologies that are used for performing the safety analyses identified in Appendix A of Enclosure I are not associated with determining TS COLR parameters.

<u>ALTERNATIVE SOURCE TERM (AST) IMPLEMENTATION</u>

See Enclosure IV of this LAR for details associated with the implementation of the AST changes.

<u>ADDITION OF TECHNICAL SPECIFICATION 3.1.9</u>

The following revisions are proposed for the Wolf Creek Generating Station (WCGS) TS in accordance with TSTF-453-T, Revision 1. The proposed TS changes will add a new LCO 3.1.9, "RCS Boron Limitations $<$ 500°F," and the associated Bases. The new LCO will require that the Reactor Coolant System (RCS) boron concentration shall be greater than the all-rods-out (ARO) critical boron concentration when the plant is operating within the following LCO 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.

If this new LCO is not met when the plant is operating within this Applicability, Required Actions will be initiated immediately, per Condition A of LCO 3.1.9, to:

- Restore the RCS boron concentration to within limit (thereby meeting the LCO), or
- Place the Rod Control System in a condition incapable of rod withdrawal (thereby eliminating the transient initiator), or
- Increase all RCS cold leg temperatures to greater than or equal to 500°F, if the plant is operating in MODE 3 at the time of Condition A entry, so that RTS trip Function 2.b of LCO 3.3.1, Power Range Neutron Flux - Low, is available to mitigate an uncontrolled RCCA bank withdrawal event postulated to occur during low power or subcritical (startup) conditions.

New Surveillance Requirement (SR) 3.1.9.1 will verify that the LCO is met every 24 hours.

The proposed TS changes will also revise the requirements for RTS trip Function 2.b, Power Range Neutron Flux - Low, in TS Table 3.3.1-1. The Applicability for 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.

New footnotes f, h, and i will be added 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 will be worded as follows:

(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 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.

New footnote (f) will divide the current MODE 2 Applicability for RTS trip Function 2.b into critical and subcritical portions. When the reactor is critical in MODE 1 below the P-10 setpoint or 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. Condition V will be similar to existing Condition E; however, the end state for the plant Condition V will also require the initiation of actions aimed at precluding an uncontrolled RCCA bank withdrawal event from occurring or providing sufficient SDM should this event occur.

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. During the subcritical portion of MODE 2, RTS trip Function 2.b performs a required function only if all RCS cold leg temperatures are greater than or equal to 500°F, and the RCS boron concentration is less than or equal to the ARO critical boron concentration, and the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

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. In the analysis of the Uncontrolled RCCA Bank Withdrawal from a Low Power or Subcritical Condition event (otherwise referred to hereafter by the abbreviation RWFS), RTS trip Function 2.b is credited to trip the reactor when all RCS cold leg temperatures are greater than or equal to 500°F, and the RCS boron concentration is less than or equal to the ARO critical boron concentration, and the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

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. Appropriate surveillance requirements have been added to demonstrate OPERABILITY of the trip Function in the revised Applicability. New Condition W, like new Condition V, will require that an inoperable channel in RTS trip Function 2.b shall be placed in the tripped condition within 72 hours. However, unlike Condition V which has a Required Action V.2.1 to be in MODE 2 with $k_{eff} < 1.0$ within the next 6 hours if the inoperable channel is not tripped within 72 hours, Condition W also applies to the upper portion of MODE 3 and a more appropriate default state has been added as new Condition X. If the Required Action and associated Completion Time of Condition W is not met, or if RTS trip Function 2.b is unavailable to provide protection for an uncontrolled RWFS event by virtue of multiple channel inoperability, the appropriate default state is to immediately initiate action to eliminate the event initiator by new Required Actions X.1.1 and X.1.2 (initiate action to fully insert all rods and place the Rod Control System in a condition incapable of rod withdrawal) or immediately initiate action to borate the RCS to such a boron concentration that the reactor will be maintained subcritical if all rods were completely withdrawn (new Required Action X.2).

Attachments II and III provide the TS markups reflecting the above changes and the retyped TS. Attachment IV provides an information-only copy of the associated TS Bases changes.

3.0 TECHNICAL EVALUATION

CORE DESIGN AND SAFETY ANALYSIS METHODOLOGY TRANSITION

WCNOC plans to transition from its current methodology for performing core design, non-LOCA and LOCA safety analyses (Post-LOCA Subcriticality and Cooling) to the NRC approved Westinghouse methodologies for performing these analyses.

Westinghouse currently holds the analysis of record (AOR) for both the WCGS Small Break (SB) and Large Break (LB) LOCA analyses; therefore, the SBLOCA and LBLOCA analyses are not included in the methodology transition effort discussed in this LAR.

For the safety analyses that were reanalyzed, they were conservatively 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. Note that even though some analyses were performed at an uprated power (representative of an MUR), the MUR conditions (i.e., NSSS power) would bound plant operation at the current rated thermal power (RTP). This license amendment request is not requesting the NRC approval of a MUR power uprate. This LAR addresses the transition to the approved Westinghouse methodologies only.

Enclosure I to this LAR contains WCAP-17658-NP, Revision 1, “Wolf Creek Generating Station Transition of Methods for Core Design and Safety Analyses – Licensing Report.” Enclosure I summarizes the analyses that were performed to confirm that the applicable acceptance criteria are met. Section 2 of Enclosure I provide the results of the safety analyses and core design efforts. Appendix A, “Safety Evaluation Report Compliance,” of Enclosure I provides a summary of NRC approved codes and methodologies that were used for the analyses. Appendix A addresses compliance with the limitations, restrictions, and conditions specified in the NRC Safety Evaluation or NRC Safety Evaluation Report for the applicable codes and methodologies.

The following Table provides a roadmap of the Westinghouse analysis codes used in each of the affected safety analyses discussed in Enclosure 1 and identifies the applicable Updated Safety Analysis Report (USAR) section associated with the use of each code.

Summary of Analysis Codes Utilized in Postulated Accident Analyses			
Subject	**Topical Report**	**Code(s)**	**Accident Analysis in Enclosure I (USAR Section)**
Non-LOCA Thermal Transients	WCAP-7908-A	FACTRAN	2.5.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition (USAR Section 15.4.1) 2.5.6 Spectrum of Rod Cluster Control Assembly Ejection Accidents (USAR Section 15.4.8)

Summary of Analysis Codes Utilized in Postulated Accident Analyses			
Subject	**Topical Report**	**Code(s)**	**Accident Analysis in Enclosure I (USAR Section)**
Non-LOCA Safety Analysis	WCAP-14882-P-A	RETRAN	2.2.1 Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature (USAR Section 15.1.1) 2.2.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow (USAR Section 15.1.2) 2.2.3 Excessive Increase in Secondary Steam Flow (USAR Section 15.1.3) 2.2.4 Inadvertent Opening of a Steam Generator Atmospheric Relief or Safety Valve (USAR Section 15.1.4) 2.2.5 Steam System Piping Failure (USAR Section 15.1.5) 2.3.1 Loss of External Electrical Load, Turbine Trip, Inadvertent Closure of Main Steam Isolation Valves, Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip (USAR Sections 15.2.2, 15.2.3, 15.2.4, and 15.2.5) 2.3.2 Loss of Non-Emergency AC Power to the Station Auxiliaries (USAR Section 15.2.6) 2.3.3 Loss of Normal Feedwater Flow (USAR Section 15.2.7) 2.3.4 Feedwater System Pipe Break (USAR Section 15.2.8) 2.4.1 Partial and Complete Loss of Forced Reactor Coolant Flow (USAR Sections 15.3.1 and 15.3.2) 2.4.2 Reactor Coolant Pump Shaft Seizure (Locked Rotor) and Shaft Break (USAR Sections 15.3.3 and 15.3.4) 2.5.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (USAR Section 15.4.2) 2.6.1 Inadvertent Operation of the Emergency Core Cooling System During Power Operation (USAR Section 15.5.1) 2.6.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (USAR Chapter 15.5.2) 2.7.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve (USAR Section 15.6.1)

Summary of Analysis Codes Utilized in Postulated Accident Analyses			
Subject	**Topical Report**	**Code(s)**	**Accident Analysis in Enclosure I (USAR Section)**
Non-LOCA Safety Analysis	WCAP-7907-P-A	LOFTRAN	2.5.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (USAR Section 15.4.2) 2.5.3 Rod Cluster Control Assembly Misoperation (USAR Section 15.4.3) 2.8 Anticipated Transients Without SCRAM (USAR Section 15.8)
Non-LOCA Thermal / Hydraulics	WCAP-11397-P-A	RTDP	2.12 Thermal and Hydraulic Design
Neutron Kinetics	WCAP-7979-P-A	TWINKLE	2.5.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition (USAR Section 15.4.1) 2.5.6 Spectrum of Rod Cluster Control Assembly Ejection Accidents (USAR Section 15.4.8)
Multi-dimensional Neutronics	WCAP-10965-P-A	ANC	2.2.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow (USAR Section 15.1.2) 2.2.4 Inadvertent Opening of a Steam Generator Atmospheric Relief or Safety Valve (USAR Section 15.1.4) 2.2.5 Steam System Piping Failure (USAR Section 15.1.5) 2.5.3 Rod Cluster Control Assembly Misoperation (USAR Section 15.4.3)
Non-LOCA Thermal / Hydraulics	WCAP-14565-P-A	VIPRE	2.2.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow (USAR Section 15.1.2) 2.2.4 Inadvertent Opening of a Steam Generator Atmospheric Relief or Safety Valve (USAR Section 15.1.4) 2.2.5 Steam System Piping Failure (USAR Section 15.1.5) 2.4.1 Partial and Complete Loss of Forced Reactor Coolant Flow (USAR Sections 15.3.1 and 15.3.2) 2.4.2 Reactor Coolant Pump Shaft Seizure (Locked Rotor) and Shaft Break (USAR Sections 15.3.3 and 15.3.4) 2.5.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition (USAR Section 15.4.1) 2.5.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (USAR Section 15.4.2)

Summary of Analysis Codes Utilized in Postulated Accident Analyses			
Subject	**Topical Report**	**Code(s)**	**Accident Analysis in Enclosure I (USAR Section)**
			2.5.3 Rod Cluster Control Assembly Misoperation (USAR Section 15.4.3) 2.12 Thermal and Hydraulic Design
Steam Generator Tube Rupture	WCAP-10698-P-A WCAP-14882-P-A	RETRAN	2.7.2 Steam Generator Tube Rupture (SGTR) - (USAR Section 15.6.3) 2.7.3 Steam Generator Tube Rupture – Input to Dose (USAR Section 15.6.3)

Regarding the impact of the issue of fuel thermal conductivity degradation (TCD) on the Westinghouse codes and methods, Westinghouse provided a discussion of the TCD impact in LTR-NRC-12-18, Letter from J. A. Gresham (Westinghouse) to USNRC Document Control Desk, "Westinghouse Response to December 16, 2011 NRC Letter Regarding Nuclear Fuel Thermal Conductivity Degradation" (Reference 2) and justified continued operation of the plants analyzed with Westinghouse codes and methods. The Westinghouse codes and methods applied in the non-LOCA analyses discussed in Enclosure I are consistent with those evaluated for TCD in Reference 2, and therefore the conclusions presented in Reference 2 are applicable to the WCGS.

The methodology transition described above results in the following Safety Limits (SLs), TS, and Specification changes:

1. SLs 2.1.1 "Reactor Core SLs," states:

   "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:

   "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."

   SLs 2.1.1.1 currently identifies the Revised Thermal Design Procedure (RTDP, which is contained in WCAP-11397-P-A, "Revised Thermal Design Procedure") design limit Departure from Nucleate Boiling Ratio (DNBR) for the WRB-2 correlation. The design limit DNBR is the basis for the 95 percent probability at a 95 percent confidence level that the limiting rod in the core will not undergo DNB during all Condition I and II transients. The RTDP design limit DNBR only serves as the DNB design basis for accidents initiating from nominal hot full power conditions; it does not serve as the DNB design basis for accidents that initiate from Hot Zero Power (HZP) conditions such as the HZP Steamline Break (SLB) and Uncontrolled Rod Cluster Controlled Assembly

(RCCA) Bank Withdrawal from Subcritical events. The DNBR limits listed in SLs 2.1.1.1 have therefore been revised to reflect the NRC approved correlation limit DNBR values for the WRB-2 correlation from WCAP-10444-P-A, "Reference Core Report – VANTAGE 5 Fuel Assembly," (Reference 3) and for the ABB-NV and WLOP correlations from WCAP-14565-P-A Addendum 2-P-A, "Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications," (Reference 4) which cover the DNB design bases for all accident analyses. The Thermal and Hydraulic design basis and methodology, the DNB methodology, and DNB Correlations and Limits are discussed in Sections 2.12.2.1, 2.12.2.1.2 and 2.12.2.1.3, respectively, of Enclosure I of this LAR.

These W-3 alternative correlations, ABB-NV and WLOP, are replacing the W-3 correlation for accidents listed in the WCGS licensing basis; therefore, the W-3 correlation is being deleted from SLs 2.1.1.1, and is being replaced with the ABB-NV and WLOP correlations.

Appendix A, Section A.5, item 1 of Enclosure I of this LAR discusses the application of the W-3 alternative correlations (ABB-NV and WLOP):

> "For conditions where WRB-2 is not applicable, analyses were performed using approved secondary CHF correlations (such as ABB-NV and WLOP) in compliance with the SER conditions licensed for use in the VIPRE code (WCAP-14565-P-A and its Addendum 2-P-A, Reference A.5-4)."

The ABB-NV correlation was specifically used for the DNB analysis of the Uncontrolled RCCA Bank Withdrawal from Subcritical event (discussed in Section 2.12.3.8 of Enclosure I) and for the DNB analysis of axial power distributions that were limiting in the fuel region below the first mixing vane grid (discussed in Section 2.12.3.2 of Enclosure I).

The WLOP correlation was used in the DNB analysis of the HZP SLB event (discussed in Section 2.12.3.6 of Enclosure I).

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

The Allowable Value for TS Table 3.3.1-1 Function 10, "Reactor Coolant Flow - Low" is:

> ≥ 88.9% of design flow - 90,324 gpm

The proposed change would revise the TS Allowable Value to:

> "≥ 88.9% of Normalized Flow"

The proposed change to TS Table 3.3.1-1 Function 10, would delete Footnote "m" which identifies the "% of design flow - 90,324 gpm" portion of the Allowable Value.

The existing WCGS Allowable Value is revised to be consistent with the assumptions of the new safety analysis methodology (the use of Normalized Flow, instead of design loop flow).

The RCS loss of flow events that credit percent of Normalized RCS flow are the Partial Loss of Forced Reactor Coolant Flow (discussed in Section 2.4.1 of Enclosure I) and

Reactor Coolant Pump (RCP) Shaft Seizure (Locked Rotor) and RCP Shaft Break (discussed in Section 2.4.2 of Enclosure I).

3. TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," LCO states:

> "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 ≥ 37.1 x 10^4 gpm and greater than or equal to the limit specified in the COLR."

The proposed change would revise the RCS minimum measured flow (MMF) specified in TS 3.4.1 LCO Part c. from 37.1 x 10^4 gpm to 376,000 gpm. The new value for MMF (376,000 gpm) would then be relocated to the COLR. The RCS flow value specified in TS 3.4.1 LCO Part c. would be replaced by the RCS TDF of 361,200 gpm. The RCS TDF flow value of 361,200 gpm would also replace the current RCS flow value of 37.1 x 10^4 gpm that is specified in SR 3.4.1.3 and SR 3.4.1.4 of TS 3.4.1.

Replacing the MMF value with the TDF value in the TS and relocating the MMF value to the COLR allows the value to be changed. However, in accordance with TS 5.6.5.d "The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC." Therefore, any changes to the value of 376,000 gpm would be provided to the NRC.

The MMF value that is specified in the COLR is revised from 37.1×10^4 gpm to 376,000 gpm to provide additional DNBR margin for the Uncontrolled RCCA Bank Withdrawal at Power non-LOCA safety analysis, which is discussed in Section 2.5.2 of Enclosure I.

The other non-LOCA safety analyses where the MMF is an input, assumed an RCS flow value of 371,000 gpm.

The non-LOCA safety analyses, where TDF is an input, assumed an RCS flow value of 361,200 gpm. The TDF value of 361,200 gpm was assumed in the following non-LOCA events:

- Feedwater system malfunctions that result in an increase in feedwater flow (zero power case), which is discussed in Section 2.2.2 of Enclosure I,
- Inadvertent opening of a steam generator atmospheric relief or safety valve, which is discussed in Section 2.2.4 of Enclosure I,
- Steam system piping failure (SLB) at zero power, which is discussed in Section 2.2.5.1 of Enclosure I,
- Loss of external electrical load, turbine trip, inadvertent closure of main steam isolation valves, and loss of condenser vacuum (peak RCS pressure

and peak MSS pressure cases), which are discussed in Section 2.3.1 of Enclosure I,

- Loss of non-emergency AC power to the station auxiliaries, which is discussed in Section 2.3.2 of Enclosure I,
- Loss of normal feedwater flow, which is discussed in Section 2.3.3 of Enclosure I,
- Feedwater system pipe break, which is discussed in Section 2.3.4 of Enclosure I,
- RCP shaft seizure (locked rotor) and RCP shaft break (peak RCS pressure / peak clad temperature case), which is discussed in Section 2.4.2 of Enclosure I,
- Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition; applied flow is a fraction of TDF corresponding to two reactor coolant loops operating, which is discussed in Section 2.5.1 of Enclosure I,
- Uncontrolled RCCA bank withdrawal at power (peak RCS pressure cases), which is discussed in Section 2.5.2 of Enclosure I,
- Spectrum of RCCA ejection accidents (full TDF for the full power cases and a fraction of TDF, corresponding to two reactor coolant loops operating, for the zero power cases), which is discussed in Section 2.5.6 of Enclosure I,
- Inadvertent operation of the ECCS during power operation, which is discussed in Section 2.6.1 of Enclosure I,
- CVCS malfunction that increases reactor coolant inventory, which is discussed in Section 2.6.2 of Enclosure I, and
- ATWS, which is discussed in Section 2.8.1 of Enclosure I.

Of the events listed above, explicit thermal-hydraulic (DNBR) analyses were performed for the SLB at zero power event, which is discussed in Section 2.12.3.6 of Enclosure I, and the Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition event, which is discussed in Section 2.12.3.8 of Enclosure I.

The NRC SE for WCAP-14483-A, “Generic Methodology for Expanded Core Operating Limits Report,” (Reference 5) discusses that the NRC approved analysis flow must be retained in the TS.

The NRC SE for WCAP-14483 states:

> “…the staff recommended 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 (e. g., maximum tube plugging) should be retained in the TS to assure that a lower flow rate than reviewed by the staff would not be used.”

Therefore, the TDF value of 361,200 gpm, which includes a maximum SG tube plugging level of 10%, is the minimum RCS flow rate that is retained in the TS to assure that a lower flow rate than that reviewed by the staff would not be used, as discussed above in the NRC SE for WCAP-14483.

4. TS 3.7.1, “Main Steam Safety Valves (MSSVs),” LCO requires 5 OPERABLE MSSVs per steam generator (SG). TS 3.7.1 Table 3.7.1-1, “OPERABLE Main Steam Safety Valves versus Maximum Allowable Power,” specifies the power limits (in % RATED THERMAL POWER (RTP)) applicable when the number of OPERABLE MSSVs per SG is less than 5. Table 3.7.1-1 specifies the following limits:

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (% RTP)
4	87
3	65
2	44

The proposed change would revise Table 3.7.1-1 as follows:

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

The standard, USAR Chapter 15 Loss of Load/Turbine Trip (LOL/TT) analysis where all MSSVs are assumed to be OPERABLE is discussed in Section 2.3.1 of Enclosure I.

In addition to this analysis, Westinghouse performed a supplementary analysis of the LOL/TT 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 TS Table 3.7.1-1, involved 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 analyses are consistent with those used in the case that considers peak Main Steam System (MSS) pressure concerns in Section 2.3.1 of Enclosure I. For each scenario of the number of OPERABLE MSSVs, the supplementary LOL/TT analysis determined the respective maximum initial power level for which the resultant peak MSS pressure satisfies the applicable safety analysis limit corresponding to 110% of the MSS design pressure.

5. Specification 5.6.5, “CORE OPERATING LIMITS REPORT (COLR),” Section b. lists the analytical methods used to determine the core operating limits.

The proposed change would delete the following WCNOC related analytical methods listed in Section b. of Specification 5.6.5:

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

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

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

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

The proposed change would add the following NRC approved Westinghouse analytical methodology to those listed in Section b. of Specification 5.6.5:

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

Due to the transition to Westinghouse Reload methodology the WCNOC Reload methodology was replaced with the Westinghouse Reload methodology listed above. The NRC approval for this Westinghouse methodology is listed below.

NRC Safety Evaluation Report dated May 28, 1985, "Acceptance for Referencing of Licensing Topical Report WCAP-9272(P)/9273(NP), Westinghouse Reload Safety Evaluation Methodology."

The NRC Safety Evaluation Report (cited above) described WCAP-9272-P-A as follows:

"This report describes the Westinghouse methodology for performing the safety evaluation of reload cores. The method assumes the existence of a valid conservative safety analysis, the reference analysis, and a set of key safety parameters for each accident or transient analyzed. The values of the input safety parameters in the reference safety analysis are selected to bound conservatively the values expected in subsequent cycles. If all reload safety parameters for a core are conservatively bounded, the reference safety analysis is assumed to be valid, and no further analysis is considered necessary. When a reload safety parameter is not bounded, further analysis is considered necessary to ensure that the required margin of safety is maintained for the accident in question. This last determination is made either through a complete reanalysis of the accident, or through a simpler, conservative quantitative evaluation process."

WCAP-9272-P-A contains the reload methodology (as described above) that is used to evaluate the reload core design for numerous plants with Westinghouse fuel assemblies.

WCAP-9272-P-A, the Westinghouse Reload Methodology, which is being added to Specification 5.6.5, is the only methodology that is associated with the determination of a TS COLR parameter.

The other NRC approved methodologies that are used for performing the safety analyses identified in Appendix A of Enclosure I are not associated with determining TS COLR parameters.

Due to the changes to Specification 5.6.5 described above, the list of COLR methodologies is re-numbered accordingly. The re-numbering of the list of methodologies in Specification 5.6.5 is an administrative change.

The addition of the analytical methods by topical report number and title is consistent with Amendment No. 144, (Reference 6). Amendment No. 144 adopted TSTF-363, "Revise Topical Report References in ITS 5.6.5, COLR," and the NRC concluded in the safety evaluation that the proposed change to only list the NRC approved methodology by topical report number and title is acceptable. Additionally, in a letter from the NRC to the TSTF (Reference 7) the NRC indicated that the NRC staff does not intend to backfit licensees that have these travelers (TSTF-363, TSTF-408 or TSTF-419) already in their TSs. The changes proposed to Specification 5.6.5 are consistent with the NRC published Revision 4 of NUREG-1431, "Standard Technical Specifications Westinghouse Plants."

FULL IMPLEMENTATION OF ALTERNATIVE SOURCE TERM

See Enclosure IV of this LAR for the details associated with the implementation of the AST changes.

ADDITION OF TECHNICAL SPECIFICATION 3.1.9

An uncontrolled RCCA bank withdrawal event is analyzed from both a subcritical and low power startup condition. In the USAR Chapter 15.4 analysis, this event is terminated by the Power Range Neutron Flux - Low trip Function. The Source Range Neutron Flux and Intermediate Range Neutron Flux trip Functions are also available to terminate an RCCA bank withdrawal from subcritical, but are not explicitly credited in the safety analysis to terminate the event.

The Power Range Neutron Flux - Low trip Function is only capable of providing protection for an 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. Additionally, although not explicitly analyzed while the plant is in MODE 3 when the RCS temperature is less than 500°F nor while the plant is in MODES 4 and 5, the Source Range Neutron Flux trip Function is implicitly credited to provide protection for an RCCA bank withdrawal event occurring from those initial conditions.

Therefore, since there is no explicit RCCA bank withdrawal analysis that is performed for MODE 3 when the RCS temperature is less than 500°F, nor for MODES 4 or 5, new LCO 3.1.9 will require that the RCS is borated to greater than the ARO critical boron concentration to provide sufficient SDM if the rods are capable of being withdrawn in these MODES. Borating the RCS to greater than the ARO critical boron concentration when the RCCA banks are capable of rod withdrawal provides sufficient SDM in the event of an uncontrolled RCCA bank withdrawal event from a subcritical condition when the RCS temperature is less than 500°F.

New LCO 3.1.9 does not cover that hypothetical portion of MODE 2 with the reactor subcritical ($k_{eff} < 1.0$) and any combination of one or both of the following specified conditions in the Applicability:

- all RCS cold leg temperatures ≥ 500°F, or
- Rod Control System incapable of rod withdrawal.

The proposed changes are more restrictive than the existing TS given the additional requirements being added in the form of new LCO 3.1.9 on boration requirements when the RCS temperature is below 500°F and by virtue of extending the Applicability of RTS trip Function 2.b, Power Range Neutron Flux - Low, to the upper portion of MODE 3 with additional Condition/Required Actions if the LCO is not met. The current Applicability for RTS trip

Function 2.b includes all of MODE 2 and invokes Condition E for an inoperable channel. The revised Applicability for RTS trip Function 2.b does not cover that limited portion of MODE 2 with the reactor subcritical ($k_{eff} < 1.0$) and any combination of one or more of the following specified conditions in the Applicability:

- any RCS cold leg temperature < 500°F, or
- RCS boron concentration greater than the ARO critical boron concentration, or
- Rod Control System incapable of rod withdrawal and all rods fully inserted.

It is extremely unlikely that the plant could remain in MODE 2 with k_{eff} between 0.99 and 1.0 with the RCS highly borated or all control and shutdown rods fully inserted and the Rod Control System disabled. During the limited portion of MODE 2 excluded by these specified conditions, protection against a positive reactivity transient is provided by virtue of new LCO 3.1.9 such that the protection afforded by RTS trip Function 2.b is not required. Correspondingly, no Condition entry for an inoperable channel in RTS trip Function 2.b is needed for this limited portion of MODE 2. The proposed breakdown of MODE 2 Applicability for RTS trip Function 2.b into critical and subcritical portions is similar to the respective Applicabilities of LCO 3.1.1 for SDM and LCO 3.1.6 for Control Bank Insertion Limits. During the subcritical portion of MODE 2, RTS trip Function 2.b performs a required function only if all RCS cold leg temperatures are greater than or equal to 500°F, and the RCS boron concentration is less than or equal to the ARO critical boron concentration, and the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

<u>CORE DESIGN AND SAFETY ANALYSIS METHODOLOGY TRANSITION</u>

The safety analyses acceptance criteria are based on meeting the relevant regulatory requirements of 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants." The GDC that form the bases of the applicable safety analysis acceptance criteria are discussed in WCAP-17658-NP, Revision 1, "Wolf Creek Generating Station Transition of Methods for Core Design and Safety Analyses – Licensing Report," provided in Enclosure I of this LAR. The following GDCs are applicable to the safety analyses discussed in Enclosure I of this LAR:

Criterion 10 – Reactor design. The 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. Instrumentation 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. The 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. The 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 25 – Protection system requirements for reactivity control malfunctions. The 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. Two 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.

Criterion 27 – Combined reactivity control systems capability. The 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. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Criterion 31 – Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Criterion 35 – Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to 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.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

FULL IMPLEMENTATION OF ALTERNATIVE SOURCE TERM

10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” Criterion 19 - Control Room. This criterion is applicable insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of allowable values.

RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” provides guidance to licensees of operating power reactors on acceptable applications of alternative source terms; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes an acceptable AST and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

NRC Generic letter 2003-01, “Control Room Habitability,” requests addressees to submit information that demonstrates that the control room at each of their respective facilities complies with the current licensing and design bases and applicable regulatory requirements, and that suitable design, maintenance and testing control measures are in place for maintaining this compliance.

RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” provides guidance on determining atmospheric relative concentration (χ/Q) values in support of design basis control room radiological habitability assessments at nuclear power plants. This document describes methods acceptable to the NRC staff for determining χ/Q values that will be used in control room radiological habitability assessments performed in support of applications for licenses and license amendment requests. Many of the regulatory positions presented in this guide represent substantial changes from procedures previously used to determine atmospheric relative concentrations for assessing the potential control room radiological consequences for a range of postulated accidental releases of radioactive material to the atmosphere. These revised procedures are largely based on the NRC sponsored computer code, ARCON96.

RG 1.145, “Atmospheric Dispersion Models For Potential Accident Consequence Assessments At Nuclear Power Plants,” provides guidance to determine relative concentrations for assessing the potential offsite radiological consequences for a range of postulated accidental releases of radioactive material to the atmosphere. These procedures include consideration of plume meander, directional dependence of dispersion conditions, and wind frequencies for various locations around actual exclusion area and low population zone (LPZ) boundaries.

ADDITION OF TECHNICAL SPECIFICATION 3.1.9

NUREG-0800, "U. S. Nuclear Regulatory Commission Standard Review Plan," Section 15.4.1, "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical or Low-Power Condition" requires that GDC-10, GDC-20, and GDC-25 be met. These criteria are met if DNBR and fuel centerline temperature limits are satisfied.

GDC-10 requires that specified acceptable fuel design limits shall not be exceeded during normal operation, including the effects of anticipated operational occurrences.

GDC-13 requires that instrumentation 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.

GDC-20 requires that the protection system(s) 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.

GDC-21 requires that the protection system(s) shall be designed for high functional reliability and testability.

GDC-22 through GDC-25 require various design attributes for the protection system(s), including independence, safe failure modes, separation from control systems, and requirements to assure that specified acceptable fuel design limits are not exceeded in the event of reactivity control malfunctions.

GDC-26, GDC-28 and GDC 29 require that the plant have two independent reactivity control systems, with at least one of the systems capable of holding the reactor core subcritical under cold conditions, and that specified acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. The maximum reactivity worth of the control and shutdown rods and the maximum rates of reactivity insertion employing the rods and boron removal are limited to values that prevent any reactivity increase from rupturing the reactor coolant system boundary or disrupting the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling. The reactivity control and protection systems are designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

10 CFR 50.55a(h) requires that the protection systems meet IEEE 279-1971. Section 4.1 of IEEE 279-1971 discusses the general functional requirement for protection systems that they automatically initiate appropriate protective action whenever a condition monitored by the system reaches a preset level, i.e., the nominal Trip Setpoint.

CONCLUSION

In conclusion, based on the considerations discussed above and detailed in the remainder of this submittal, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the requested license amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.2 Significant Hazards Consideration

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Wolf Creek Nuclear Operating Corporation (WCNOC) hereby requests an amendment to Renewed Facility Operating License No. NPF-42 for the Wolf Creek Generating Station (WCGS).

The proposed amendment request revises Safety Limits (SLs) 2.1.1, "Reactor Core SLs," Technical Specification (TS) TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," TS 3.7.1, "Main Steam Safety Valves (MSSVs)," and Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)" to replace the existing analysis methodologies with standard Westinghouse developed and NRC approved analysis methodologies.

In addition, the proposed amendment request revises the TS definitions of DOSE EQUIVALENT I-131, and DOSE EQUIVALENT XE-133, TS 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation," Specification 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," and 5.5.18, "Control Room Envelope Habitability Program," to revise the WCGS licensing basis by adopting the Alternative Source Term (AST) radiological analysis methodology as allowed by 10 CFR 50.67, "Accident source term." This amendment request represents a full scope implementation of the AST as described in NRC RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0.

Further, the proposed amendment will add a new Tech Spec 3.1.9, RCS Boron Limitations < 500 °F and the associated Bases. The proposed changes will also revise the Applicability of Function 2. b., Power Range Neutron Flux- Low in Table 3.3.1-1 and add a new Applicability, new Conditions T and U, and the appropriate SRs required to demonstrate Operability of the Function. New Footnotes c, d, and e are added to the Applicability of Function 2. b., Power Range Neutron Flux- Low in Table 3.3.1-1 are added to address the revised Applicability of the Function. These changes are required to reflect the current safety analysis assumptions regarding the RCCA bank withdrawal from subcritical event.

WCNOC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," Part 50.92(c) as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The new core design, non-loss-of-coolant-accident (non-LOCA) and Post-LOCA Subcriticality and Cooling analyses and resulting TS changes will continue to ensure the applicable safety limits are not exceeded during any conditions of normal operation, for design basis accidents (DBAs) as well as any Anticipated Operational Occurrence (AOO). The methods used to perform the affected safety analyses are based on methods previously found acceptable by the NRC and conform to applicable regulatory guidance. Application of these NRC approved methods will continue to ensure that acceptable operating limits are established to protect the integrity of the Reactor Coolant System (RCS) and fuel cladding during normal operation, DBAs, and any AOOs. The requested TS changes proposed to conform to the new methodologies do not involve any operational changes that could affect system reliability, performance, or the possibility of operator error. The proposed changes do not affect any postulated accident precursors, or accident mitigation systems, and do not introduce any new accident initiation mechanisms.

Adoptions of the AST and pursuant TS changes and the changes to the atmospheric dispersion factors have no impact to the initiation of DBAs. Once the occurrence of an accident has been postulated, the new accident source term and atmospheric dispersion factors are an input to analyses that evaluate the radiological consequences. The proposed changes do not involve a revision to the design or manner in which the facility is operated that could increase the probability of an accident previously evaluated in Chapter 15 of the Updated Safety Analysis Report (USAR).

The structures, systems and components affected by the proposed changes act to mitigate the consequences of accidents. Based on the AST analyses, the proposed changes do revise certain performance requirements; however, the proposed changes do not involve a revision to the parameters or conditions that could contribute to the initiation of an accident previously discussed in Chapter 15 of the USAR. Plant specific radiological analyses have been performed using the AST methodology and new atmospheric dispersion factors. Based on the results of these analyses, it has been demonstrated that the control room dose consequences of the limiting events considered in the analyses meet the regulatory guidance provided for use with the AST, and the offsite doses are within acceptable limits. This guidance is presented in 10 CFR 50.67 and RG 1.183.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any previously evaluated?

Response: No

Implementation of the new core design, non-loss-of-coolant-accident (non-LOCA) and Post-LOCA Subcriticality and Cooling analyses and resulting TS changes do not alter or involve any design basis accident initiators and do not involve a physical alteration of the plant (no new or different type of equipment will be installed). The proposed change does not adversely affect the design function or mode of operations of structures, systems and components in the facility important to safety. The structures, systems and components important to safety will continue to operate in the same manner as before, therefore, no new failure modes are created by this proposed change. As such, the proposed change does not create any new failure modes for existing equipment or any new limiting single failures. Additionally the proposed change does not involve a change in the methods governing normal plant operation and all safety functions will continue to perform as previously assumed in accident analyses. Thus, the proposed change does not adversely affect the design function or operation of any structures, systems, and components important to safety. The proposed change does not involve changing any accident initiators.

Implementation of AST and the associated proposed TS changes and new atmospheric dispersion factors do not alter or involve any design basis accident initiators. A design modification will be implemented in support of the proposed AST change that will eliminate the need for local operator action to isolate a failed CREVS train. The proposed change does not adversely affect the design function or mode of operations of structures, systems and components in the facility important to safety. The structures, systems and components important to safety will continue to function in the same manner as before after the AST is implemented. Therefore, no new failure modes are created by this proposed change.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

The proposed methodology and TS changes will not adversely affect the operation of plant equipment or the function of equipment assumed in the accident analysis. The proposed changes do not adversely affect the design and performance of the structures, systems, and components important to safety. Therefore, the required safety functions will continue to be performed consistent with the assumptions of the applicable safety analyses. In addition, operation in accordance with the proposed TS change will continue to ensure that the previously evaluated accidents will be mitigated as analyzed. The NRC approved safety analysis methodologies include restrictions on the choice of inputs, the degree of conservatism inherent in the calculations, and specified event acceptance criteria. Analyses performed in accordance with these methodologies will not result in adverse effects on the regulated margin of safety. As such, there is no significant reduction in a margin of safety.

The results of the AST analyses are subject to the acceptance criteria in 10 CFR 50.67. The analyzed events have been carefully selected, and the analyses supporting these changes have been performed using approved methodologies to ensure that analyzed events are bounding and safety margin has not been reduced. The dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67 and RG 1.183. Thus, by meeting the applicable regulatory limits for AST, there is no significant reduction in a margin of safety. New control room atmospheric dispersion factors (χ/Qs) based on site specific meteorological data, calculated in accordance with the guidance of RG 1.194, utilizes more recent data and improved calculation methodologies.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

4.3 Conclusion

Based on the considerations discussed above, 1) there is a reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, 2) such activities will be conducted in compliance with the NRC'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.

4.4 Application of WCAP-17504-P-A

Enclosure II of this LAR discusses the application of WCAP-17504-P-A (Reference 15). The NRC Safety Evaluation approved the use of the WCAP-17504 methodology with the following condition:

"*As described above in Section 3.1.8, for any LARs to implement WCAP-17504-P/WCAP-17504-NP, Revision 1, for plants with non-Westinghouse NSSS vendor specified equipment, the licensee should state whether it has confirmed with the individual equipment vendors that the reference accuracy, drift, and other instrument channel component performance uncertainties have been estimated at the 95/95 two-sided statistical level. If the licensee has not been able to confirm whether the data was presented as 95/95 data, then the staff shall audit the licensee's data analysis to verify the licensee (or Westinghouse, on behalf of the licensee) has appropriately adjusted the available raw vendor data so that it is representative of high confidence (i.e., 95/95) tolerance interval information.***"**

Westinghouse, acting on behalf of WCNOC, was not able to confirm with the non-Westinghouse equipment vendors that their design specifications such as reference accuracy, drift, and other instrument performance uncertainties have been estimated at the 95/95 two-sided statistical level. Accordingly, when vendor or plant performance data was available, additional data analyses were performed or, the vendor specifications have been adjusted so that terms used in the uncertainty analysis represent a high confidence (i.e., 95/95) tolerance interval. These analyses will be made available for audit upon request by the NRC.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. WCAP-9272-P-A, Revision 0, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

2. Letter from J. A. Gresham Westinghouse to USNRC Document Control Desk, LTR-NRC-12-18 "Westinghouse Response to December 16, 2011 NRC Letter Regarding Nuclear Fuel Thermal Conductivity Degradation (TAC No. ME5186) (Proprietary)," February 17, 2012. ADAMS Accession No. ML12053A105.

3. WCAP-10444-P-A, "Reference Core Report – VANTAGE 5 Fuel Assembly," September 1985.

4. WCAP-14565-P-A Addendum 2-P-A, “Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications,” April 2008.

5. WCAP-14483-A, “Generic Methodology for Expanded Core Operating Limits Report,” January 1999.

6. Letter from J. N. Donohew, USNRC, to O. L. Maynard, WCNOC, “Wolf Creek Generating Station – Issuance of Ammendment Re: Relocation of Cycle Specific Parameters to the CORE OPERATING LIMITS REPORT (TAC No. MB1638),” March 28, 2003. ADAMS Accession No. ML-020180190.

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