WOLF CREEK
NUCLEAR OPERATING CORPORATION

John P. Broschak
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August 13, 2013
ET 13-0023

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

Reference: NRC Memorandum dated November 1, 2012, "Summary of September 20, 2012, Category 1 Meeting to Discuss Licensee's Plans to Transition from Current Methodologies to NRC-Approved Westinghouse Methodologies for Performing the Core Design and the Non-Loss-of-Coolant Accident Analyses at Wolf Creek (TAC NO. ME9495)" ADAMS Accession No. ML12276A265

Subject: Docket No. 50-482: License Amendment Request for the Transition to Westinghouse Core Design and Safety Analysis

Gentlemen:

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 2.1.1, "Reactor Core SLs," Technical Specification (TS) 3.3.1, "Reactor Trip System (RTS) Instrumentation," TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start 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 WCNOC methodology for performing core design, non-loss-of-coolant-accident (non-LOCA) and LOCA safety analyses (for Post-LOCA Subcriticality and Cooling only) with standard Westinghouse developed and Nuclear Regulatory Commission (NRC) approved analysis methodologies. As part of the transition to the generic Westinghouse NRC approved methodologies, instrumentation setpoint and control uncertainty calculations were performed based on the current Westinghouse Setpoint Methodology. This amendment request also includes the adoption of Option A of Technical Specification Task Force (TSTF) TSTF-493-A, Revision 4, "Clarify Application of Setpoint Methodology for LSSS Functions."

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In addition, the proposed amendment request revises the TS definitions of DOSE EQUIVALENT I-131, and DOSE EQUIVALENT XE-133, and Specification 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," 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 represents a full scope implementation of the AST as described in NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0. In conjunction with the full scope implementation of the AST, the proposed amendment request includes changes to adopt TSTF-51-A, Revision 2, "Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations." The adoption of TSTF-51-A results in changes to TS 3.3.6, "Containment Purge Isolation Instrumentation," TS 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation," TS 3.3.8, "Emergency Exhaust System (EES) Actuation Instrumentation," TS 3.7.10, "Control Room Emergency Ventilation System (CREVS)," TS 3.7.11, "Control Room Air Conditioning System (CRACS)," TS 3.7.13, "Emergency Exhaust System (EES)," and TS 3.9.4, "Containment Penetrations."

The proposed change for full scope implementation of the AST and the adoption of TSTF-51-A, are fully discussed, including the associated TS, TS Bases, and Updated Safety Analysis Report (USAR) markups, in Enclosure VI of this amendment request. These proposed changes are provided in a separate enclosure to allow the implementation of the AST and TSTF-51-A to be reviewed separately from the other changes proposed in this amendment request. Enclosure VII is a CD-ROM containing meteorological data used to determine offsite, control room, and technical support center atmospheric dispersion factors.

The transition to Westinghouse core design and safety analyses methodologies was performed to utilize current methodologies and provide more consistent alignment with the Westinghouse fleet of plants. The transition also provides the corrective actions relating to WCGS non-conforming conditions associated with operator response times, assumed initial conditions, and instrumentation setpoint and control calculations. The reanalyzed Updated Safety Analysis Report (USAR) Chapter 15 transients and accidents impacted the RTS, ESFAS, and LOP DG Start instrumentation Allowable Valves and Trip Setpoints described in the current WCGS TS 3.3.1, TS 3.3.2, and TS 3.3.5. The need to revise these TSs led to the adoption of the NRC-approved TSTF-493-A, Revision 4, in order to address concerns associated with the evaluation of instrument performance. Additionally, the reanalyzed USAR Chapter 15 accidents affected the associated radioactive steam releases which required the radiological consequence analyses to be updated. The radiological consequence analyses were performed using the updated accident source term consistent with 10 CFR 50.67. Therefore, the interdependent relationships between the reanalyzed USAR Chapter 15 accidents and transients, trip setponts, and radiological consequence analyses necessitates combining these analyses and calculations into a single license amendment request.

Attachments I through V provide the Evaluation, Markup of TSs, Retyped TS pages, proposed TS Bases changes (for information only), and proposed COLR changes (for information only), respectively, in support of this amendment request. Attachments IV and V, are provided for information only. Final TS Bases changes will be implemented pursuant to TS 5.5.14, "Technical Specification (TS) Bases Control Program," at the time the amendment is implemented.

Enclosure II provides the proprietary Westinghouse Electric Company LLC, WCAP-17746-P, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station." Enclosure III provides the non-proprietary Westinghouse Electric Company LLC, WCAP-17746-NP, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station." Enclosure IV provides the proprietary Westinghouse Electric Company LLC, WCAP-17602-P, "Westinghouse Setpoint Calculations for the Wolf Creek Generating Station Control, Protection, and Indication Systems." Enclosure V provides the non-proprietary Westinghouse Electric Company LLC, WCAP-17602-NP, "Westinghouse Setpoint Calculations for the Wolf Creek Generating Station Control, Protection, and Indication Systems." As Enclosures II and IV contain information that is proprietary to Westinghouse Electric Company LLC, it is supported by an affidavit signed by Westinghouse Electric Company LLC, the owner of the information. Enclosures VIII and Enclosure IX are the Westinghouse Application for Withholding Proprietary Information from Public Disclosure, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice. As Enclosure II and Enclosure IV contain information proprietary to Westinghouse Electric Company LLC, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92, "Issuance of amendment." Pursuant to 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible or otherwise not requiring environmental review," Section (b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

The Plant Safety Review Committee reviewed this amendment application. In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," a copy of this amendment application, with attachments, is being provided to the designated Kansas State official.

On September 20, 2012, WCNOC and Westinghouse personnel conducted a pre-submittal meeting with the NRC staff to provide advance notification of plans to prepare and submit a license amendment request for the transition to the Westinghouse core design and safety analysis methodologies. The Reference documented the results of the September 20, 2012 pre-submittal meeting. A subsequent pre-submittal meeting was held on July 30, 2013.

WCNOC is requesting approval of the proposed license amendment by December 15, 2014 to support Cycle 21 operation (startup from Refueling Outage 20). Refueling Outage 20 is currently scheduled to begin in early January 2015. It is anticipated that the license amendment, as approved, will become effective upon issuance and will be implemented prior to startup from Refueling Outage 20. Activities required to support implementation of this amendment will be occurring throughout Cycle 20 and will impact the core reload design process, procedures, and training programs. Approval of this license amendment request beyond the requested date may impact the ability to restart the plant from Refueling Outage 20 as currently scheduled.

There are no regulatory commitments contained in this submittal. If you have any questions concerning this matter, please contact me at (620) 364-4085, or Mr. Michael Westman at (620) 364-4009.

Sincerely,

John P. Broschak

JPB/rlt

Attachments:
- I Evaluation
- II Proposed Technical Specification Changes (Mark-up)
- III Revised Technical Specification Pages
- IV Proposed TS Bases Changes (for information only)
- V Proposed COLR Changes (for information only)

Enclosures:
- I WCAP-17658-NP, Revision 0, "Wolf Creek Generating Station Transition of Methods for Core Design and Safety Analyses – Licensing Report" (Non-Proprietary)
- II WCAP-17746-P, Revision 0, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station" (Proprietary)
- III WCAP-17746-NP, Revision 0, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station" (Non-Proprietary)
- IV WCAP-17602-P, Revision 0, "Westinghouse Setpoint Calculations for the Wolf Creek Generating Station Control, Protection, and Indication Systems" (Proprietary)
- V WCAP-17602-NP, Revision 0, "Westinghouse Setpoint Calculations for the Wolf Creek Generating Station Control, Protection, and Indication Systems" (Non-Proprietary)
- VI Full Scope Implementation of Alternative Source Term (Non-Proprietary)
- VII CD-ROM containing Meteorological Data Used to Determine Offsite, Control Room and TSC Atmospheric Dispersion Factors
- VIII CAW-13-3766, "Application for Withholding Proprietary Information from Public Disclosure and Affidavit for WCAP-17446-P, Revision 0, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station" (Proprietary)"
- IX CAW-13-3767, "Application for Withholding Proprietary Information from Public Disclosure and Affidavit for WCAP-17446-P, Revision 0, "Westinghouse Setpoint Calculations for the Wolf Creek Generating Station Control, Protection, and Indication Systems" (Proprietary)"

cc: T. A. Conley (KDHE), w/a, w/e (Non-Proprietary only)
C. F. Lyon (NRC), w/a, w/e
N. F. O'Keefe (NRC), w/a, w/e
S. A. Reynolds (NRC), w/a, w/e
Senior Resident Inspector (NRC), w/a, w/e

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

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

By John P. Broschak
John P. Broschak
Vice President Engineering

SUBSCRIBED and sworn to before me this 13th day of August, 2013.

Gayle Shepheard
Notary Public

GAYLE SHEPHEARD
Notary Public - State of Kansas
My Appt. Expires 7/24/2015

Expiration Date 7/24/2015

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,"
    - 3.3.1, "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. Replace the existing WCNOC Setpoint Methodology (that was developed by WCNOC) with the standard Westinghouse Setpoint Methodology. This proposed change would result in revisions to the following Technical Specifications (TS):

    - 3.3.1, "Reactor Trip System (RTS) Instrumentation,"
    - 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," and
    - 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation."

3. The adoption of Option A of Technical Specification Task Force (TSTF)-493-A, Revision 4, "Clarify Application of Setpoint Methodology for LSSS Functions." This proposed change would result in revisions to the following TS:

    - 3.3.1, "Reactor Trip System (RTS) Instrumentation," and
    - 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

    In addition to the TS listed above, the adoption of TSTF-493-A requires Bases changes for the following TS:

    - 3.3.1, "Reactor Trip System (RTS) Instrumentation,"
    - 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation,"
    - 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation,"
    - 3.3.6, "Containment Purge Isolation Instrumentation,"

- 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation," and
- 3.3.8, "Emergency Exhaust System (EES) Actuation Instrumentation."

4. 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 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0. Included with the adoption of AST radiological analysis methodology, the proposed changes include the adoption of TSTF-51-A, Revision 2, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations." Enclosure VI of this amendment request contains a complete description of the proposed AST and TSTF-51-A 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 VI) to facilitate a separate review of the AST changes, independent from the other proposed changes in this amendment request.

2.0 DETAILED DESCRIPTION

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, "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 Trip Setpoint of TS 3.3.1, Function 10, "Reactor Coolant Flow - Low" (identified in the current TS 3.3.1 Bases Table B 3.3.1-1) is:

   "≥ 89.9% of loop design flow (90,324 gpm)"

   The proposed change would revise the Trip Setpoint to a Nominal Trip Setpoint (NTSP) for this RTS Function to:

   "90% of indicated loop flow"

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 3.7.10 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.

SETPOINT METHODOLOGY TRANSITION

The transition from the existing WCNOC Setpoint Methodology (that was developed by WCNOC) to the standard Westinghouse Setpoint Methodology affects several TS and the values specified in those TS. The Westinghouse Setpoint Methodology, WCAP-17746-P, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station," (provided in Enclosure II of this LAR) uses the Nominal Trip Setpoint (NTS) instead of the Allowable Value in the TS. When discussing the Nominal Trip Setpoint in this LAR the acronym NTSP will be used hereafter. The current WCNOC Setpoint Methodology uses the Allowable Value in the TS. As the Westinghouse Setpoint Methodology uses the NTSP, the primary effect on the WCGS TS due to implementing the Westinghouse methodology is the replacement of the existing WCGS TS Allowable Values with NTSPs. The implementation of the Westinghouse Setpoint Methodology only resulted in two changes to the existing WCGS Trip Setpoints. The two NTSPs proposed for use in the TS that are different than the existing WCGS Trip Setpoints are discussed below (TS 3.3.1, RTS Function 10, and TS 3.3.5, the degraded voltage Function).

The transition to the Westinghouse Setpoint Methodology affects the following TSs:

- 3.3.1, "Reactor Trip System (RTS) Instrumentation,"
- 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," and
- 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation."

The existing WCGS Trip Setpoints are listed in the Bases of TS 3.3.1 and TS 3.3.2, and are included in the SRs of TS 3.3.5.

The changes to the current instrumentation Function Allowable Value for each affected TS are identified below.

1. TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," Table 3.3.1-1 changes to replace the Allowable Values with the NTSP. The WCGS Table 3.3.1-1 uses a "single column" format specifying the Allowable Value. As such, the Allowable Value column header on RTS Table 3.3.1-1 (including Footnote (a) which states that the Allowable Value defines the Limiting Safety System Setting) would be replaced with "Nominal Trip Setpoint," without a footnote. Existing Footnote (b) on Table 3.3.1-1 would be re-lettered to Footnote (a).

   Note that some RTS Functions listed in TS 3.3.1 Table 3.3.1-1 do not have specified Allowable Values (e.g., "NA" is listed as the Allowable Value). RTS Functions without Allowable Values are not affected by the transition to the Westinghouse Methodology. Therefore, the list below only contains those RTS Functions with specified Allowable Values that would be replaced due to the transition to the setpoint methodology in WCAP-17746-P, except as noted below for RTS Function 10.

RTS Function		Existing Allowable Value	NTSP
2.a	Power Range Neutron Flux High	≤ 112.3% RTP	109% RTP
2.b	Power Range Neutron Flux Low	≤ 28.3% RTP	25% RTP
3.a	Power Range Neutron Flux High Positive Rate	≤ 6.3% RTP with time constant ≥ 2 sec	4% RTP with time constant ≥ 2 sec
3.b	Power Range Neutron Flux High Negative Rate	≤ 6.3% RTP with time constant ≥ 2 sec	4% RTP with time constant ≥ 2 sec
4.	Intermediate Range Neutron Flux	≤ 35.3% RTP	25% RTP
5.	Source Range Neutron Flux	≤ 1.6 E5 cps	10^5 cps
6.	Overtemperature ΔT	Refer to Note 1	Refer to Note 1 [(a)]
7.	Overpower ΔT	Refer to Note 2	Refer to Note 2 [(a)]
8.a	Pressurizer Pressure Low	≥ 1930 psig	1940 psig
8.b	Pressurizer Pressure High	≤ 2395 psig	2385 psig
9.	Pressurizer Water Level High	≤ 93.9% of Instrument Span	92% of instrument span
10.	Reactor Coolant Flow - Low	≥ 88.9% of design flow – 90,324 gpm	90% of indicated loop flow [(b)]

RTS Function	Existing Allowable Value	NTSP
12. Undervoltage RCPs	≥ 10355 Vac	10578 Vac
13. Underfrequency RCPs	≥ 57.1 Hz	57.2 Hz
14. SG Water Level Low-Low	≥ 22.3% of Narrow Range Instrument Span	23.5% of Narrow Range Instrument Span
16.a Turbine Trip Low Fluid Oil Pressure	≥ 534.20 psig	590.0 psig
16.b Turbine Trip Turbine Stop Valve Closure	≥ 1% open	≥ 1% open
18.a Intermediate Range Neutron Flux, P-6 Interlock	≥ 6E-11 amp	1.0E-10 amps
18.c Power Range Neutron Flux, P-8 Interlock	≤ 51.3% RTP	48% RTP
18.d Power Range Neutron Flux, P-9 Interlock	≤ 53.3% RTP	50% RTP
18.e Power Range Neutron Flux, P-10 Interlock	≥ 6.7% RTP and ≤ 13.3% RTP	10% RTP
18.f Turbine Impulse Pressure, P-13 Interlock	≤ 12.4% turbine power	10% turbine power

Table Notes:

(a) The changes to the Overtemperature ΔT and Overpower ΔT Table 3.3.1-1, Notes 1 and 2 are discussed below.

(b) Changes to the Reactor Coolant Flow - Low RTS Function include the deletion of the associated footnote (m) which specified a part of the Allowable Value (i.e., % of design flow – 90,324 gpm). The change also includes a revision to the NTSP value for the following reasons: The existing WCGS Trip Setpoint (≥ 89.9% of loop design flow (90,324 gpm) provided in the Bases of TS 3.3.1) was revised due to human factors (i.e., rounding the indicated flow value up to the next whole number of 90%) and to be consistent with the assumptions of the new safety analysis methodology (the use of indicated loop flow, instead of design loop flow). The change in the existing NTSP due to the safety analysis is discussed in Section 3.0 as part of the Core Design and Safety Analysis Methodology Transition changes.

Changes to Table 3.3.1-1 Note 1: Overtemperature ΔT

The following statement in Note 1 specifying the Overtemperature ΔT Allowable Value limit would be deleted.

> "The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.3% of ΔT span."

The description of ΔT_0 would be revised as follows:

From: ΔT_0 is the indicated ΔT at RTP, °F
To: ΔT_0 is the indicated loop ΔT at RTP, °F

The description of T' would be revised as follows:

From: T' is the nominal T_{avg} at RTP, ≤ *
To: T' is the nominal T_{avg} (T_{ref} from Rod Control) at RTP, ≤ *

Changes to Table 3.3.1-1 Note 2: Overpower ΔT

The following statement in Note 2 specifying the Overpower ΔT Allowable Value limit would be deleted.

"The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 2.6% of ΔT span."

The description of ΔT_0 would be revised as follows:

From: ΔT_0 is the indicated ΔT at RTP, °F
To: ΔT_0 is the indicated loop ΔT at RTP, °F

The description of T" would be revised as follows:

From: T" is the indicated T_{avg} at RTP (Calibration temperature for ΔT instrumentation), ≤ * °F
To T" is the nominal T_{avg} (T_{ref} from Rod Control) at RTP, ≤ * °F

The changes listed above for the Overtemperature and Overpower ΔT Notes change values (e.g., "indicated" to "nominal") or add clarifications to values (e.g., "T_{ref} from Rod Control") that are consistent with values used in WCAP-17746-P and are consistent with the non-LOCA safety analysis where T' and T" are set to the T_{ref} value.

2. TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," Table 3.3.2-1 changes to replace the Allowable Values. The WCGS Table 3.3.2-1 uses a "single column" format specifying the Allowable Value. As such, the Allowable Value column header on ESFAS Table 3.3.2-1 (including Footnote (a) which states that the Allowable Value defines the Limiting Safety System Setting) would be replaced with "Nominal Trip Setpoint" without a footnote. Footnote (b) on Table 3.3.2-1 would be re-lettered to Footnote (a).

Note that some ESFAS Functions listed in TS 3.3.2 Table 3.3.2-1 do not have specified Allowable Values (e.g., "NA" is listed as the Allowable Value). ESFAS Functions without Allowable Values are not affected by the transition to the Westinghouse Methodology. Therefore, the list below only contains those ESFAS Functions with specified Allowable Values that would be replaced due to the transition to the setpoint methodology in WCAP-17746-P.

ESFAS Function		Existing Allowable Value	NTSP
1.c	Safety Injection: Containment Pressure – High 1	≤ 4.5 psig	3.5 psig
1.d	Safety Injection: Pressurizer Pressure – Low	≥ 1820 psig	1830 psig
1.e	Safety Injection: Steam Line Pressure Low	≥ 571 psig	615 psig [(a)]
2.c	Containment Spray: Containment Pressure High – 3	≤ 28.3 psig	27 psig
3.b.(3)	Containment Phase B Isolation: Containment Pressure – High 3	≤ 28.3 psig	27 psig
4.d	Steam Line Isolation: Containment Pressure – High 2	≤ 18.3 psig	17 psig
4.e.(1)	Steam Line Isolation: Steam Line Pressure Low	≥ 571 psig	615 psig [(a)]
4.e.(2)	Steam Line Isolation: Negative Rate - High	≤ 125 psi	100 psi [(a)]
5.c	Turbine Trip and Feedwater Isolation: SG Water Level – High High (P-14)	≤ 79.7% of Narrow Range Instrument Span	78% of Narrow Range Instrument Span
6.d	Auxiliary Feedwater: SG Water Level Low – Low	≥ 22.3% of Narrow Range Instrument Span	23.5% of Narrow Range Instrument Span
6.h	Auxiliary Feedwater: Auxiliary Feedwater Pump Suction Transfer on Suction Pressure – Low	≥ 20.53 psia	21.6 psia
7.b	Automatic Switchover to Containment Sump: Refueling Water Storage Tank (RWST) Level – Low Low	≥ 35.5% of instrument span	36% of instrument span
8.b	ESFAS Interlocks: Pressurizer Pressure, P-11	≤ 1979 psig	1970 psig

Table Notes:

(a) The text of the footnotes associated with the Allowable Values for each of these ESFAS Functions is not changed, and continues to be applicable to the NTSP. Some footnotes are re-lettered (e.g., from (c) to (d)) due to the deletion and addition of footnotes discussed elsewhere in this LAR.

3. TS 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation," changes to replace the Allowable Values. The Allowable Values and NTSPs associated with TS 3.3.5 are specified in SR 3.3.5.3 (the CHANNEL CALIBRATION SR).

The current SR 3.3.5.3 states:

"Perform CHANNEL CALIBRATION with nominal Trip Setpoint and Allowable Value as follows:

a. Loss of voltage Allowable Value ≥ 82.5V, 120V bus with a time delay of 1.0 + 0.2, -0.5 sec.

Loss of voltage nominal Trip Setpoint 83V, 120V bus with a time delay of 1.0 sec.

b. Degraded voltage Allowable Value ≥ 105.9V, 120V bus with a time delay of 119 ± 11.6 sec.

Degraded voltage nominal Trip Setpoint 106.9V, 120V bus with a time delay of 119 sec."

SR 3.3.5.3 would be revised as follows:

"Perform CHANNEL CALIBRATION as follows:

a. Loss of voltage Nominal Trip Setpoint 83V, 120V bus with a time delay of 1.0 sec.

b. Degraded voltage Nominal Trip Setpoint 108.7V, 120V bus with a time delay of 119 sec."

The WCGS NTSP for the degraded voltage Function (106.9V) is revised to 108.7V to be consistent with setpoint methodology in WCAP-17746-P (Enclosure II).

IMPLEMENTATION OF TSTF-493-A, Revision 4

The proposed amendment would revise the TS by applying additional testing requirements to applicable instrumentation Functions, listed in Technical Specification Task Force (TSTF) Improved Standard Technical Specifications (STS) Change Traveler TSTF-493-A, Revision 4, "Clarify Application of Setpoint Methodology for LSSS Functions," Attachment A, "Identification of Instrument Functions to be Annotated with the TSTF-493 footnotes." TSTF-493-A Attachment A contains Functions related to those variables that have a significant safety function, as defined in Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36(c)(1)(ii)(A), thereby ensuring instrumentation will function as required to initiate protective systems or actuate mitigating systems at values equal to or more conservative than the point assumed in applicable safety analyses. These TS changes are made by the addition of individual SR footnotes to applicable instrumentation Functions in accordance with Option A of TSTF-493-A, Revision 4.

The proposed change is consistent with Option A of NRC approved Revision 4 to TSTF-493-A. The availability of this TS improvement was announced in the Federal Register on May 11, 2010 (75 FR 26294).

WCNOC proposes to add the TS SR footnotes in TSTF-493-A, Revision 4, Option A, with changes to two setpoint values to the WCGS instrumentation Functions. The changes to the two WCGS setpoint values are discussed in the Setpoint Methodology Transition section above.

WCNOC has reviewed the model safety evaluation (SE) referenced in the Federal Register Notice of Availability published on May 11, 2010 (75 FR 26294). As described herein, WCNOC has concluded that the justifications presented in TSTF-493-A, Revision 4, Option A, and the model SE prepared by the NRC staff for Option A are applicable to WCGS and support these changes to the WCGS TS.

WCNOC is proposing a variation from the TS changes described in TSTF-493-A, Revision 4 or the NRC staff's model SE referenced in the Notice of Availability.

The proposed WCGS instrumentation TS only specify a single value, the value for the NTSP instead of the Allowable Value. Although, the markups associated with TSTF-493-A do not show the option to specify only the NTSP, the text in the TSTF recognizes the potential plant specific use of NTSP as the sole value in the TS. Specifically, the TSTF discusses the application of footnotes (b) and (c) and the different types of TS Tables (i.e., single and multi-column formats). Regarding the single column format (i.e., WCGS TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," and TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation") the TSTF states: "For plants that specify the [NTSP] or [LTSP] instead of the Allowable Value, the same restrictions apply and the identification of the [LTSP] or [NTSP] in the last sentence in Note 2 is not required." In addition, the content of the TSTF footnotes (b) and (c) are also discussed in the Reviewer's Note included in the TSTF-493-A Bases markup. An example of the Bases Reviewer's Note provided by TSTF-493-A states: "The bracketed section '[NTSP and the]' of the sentence in Note (c) in Table 3.3.1-1 is not required in plant specific Technical Specifications which include a Nominal Trip Setpoint column in Table 3.3.1-1." The explanation of the use of NTSP in TSTF-493-A would be applicable to plants like WCGS that propose the use of the Westinghouse Setpoint Methodology, with only the NTSP value specified in the TS.

Additionally, Regulatory Guide 1.105, "Setpoints For Safety-Related Instrumentation," (Reference 2) which "describes a method acceptable to the NRC staff for complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the technical specification limits," states the following (bold font added for emphasis):

> "Section 4.3 of ISA-S67.04-1994 states that the limiting safety system setting (LSSS) may be the trip setpoint, an allowable value, or both. For the standard technical specifications, the staff designated the allowable value as the LSSS. In association with the trip setpoint and limiting conditions for operation (LCOs), the LSSS establishes the threshold for protective system action to prevent acceptable limits being exceeded during design basis accidents. The LSSS therefore ensures that automatic protective action will correct the abnormal situation before a safety limit is exceeded. **A licensee, with justification, may propose an alternative LSSS based on its particular setpoint methodology or license.**"

The proposed NTSP TS values calculated in accordance with WCAP-17746-P (Enclosure II of this LAR) meet the definition of an Limiting Safety System Setting (LSSS) as described in Regulatory Guide 1.105 (stated above) and as required in the TS by 10 CFR 50.36(c)(1)(ii)(A).

Further justification and details regarding the proposed WCGS NTSP values and WCAP-17746-P and WCAP-17602-P (Enclosure IV of this LAR) are discussed in Section 3.0, "Technical Evaluation" of this Attachment, under the heading "Setpoint Methodology Transition."

The implementation of TSTF-493-A, Option A consists of the addition of two footnotes to the CHANNEL CALIBRATION, CHANNEL OPERATIONAL TEST (COT), and TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) (with setpoint verification) SRs for the instrument "Functions related to those variables that have a significant safety function, as defined in Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36(c)(1)(ii)(A)." The TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," and TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," instrumentation Functions affected by the implementation of TSTF-493-A are evaluated using the applicable Attachment A of TSTF-493-A in the Technical Evaluation section of this LAR (Section 3.0). The following two TSTF-493-A footnotes would be added to the WCGS TS:

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and the as-left tolerances is specified in WCAP-17746-P.

WCAP-17746-P, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station," contains the methodology used to determine the as-found and as-left tolerances associated with each NTSP. Upon implementation of the proposed changes in this LAR, the Wolf Creek Updated Safety Analysis Report (USAR) would be updated to discuss and reference WCAP-17746-P as required by TSTF-493-A.

In WCGS TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," Table 3.3.1-1, Footnotes (b) and (c) are added to the following SRs for the RTS Functions that meet the requirements of TSTF-493-A:

- SR 3.3.1.7, Perform a COT.
- SR 3.3.1.8, Perform a COT.
- SR 3.3.1.10, Perform a CHANNEL CALIBRATION.
- SR 3.3.1.11, Perform a CHANNEL CALIBRATION.

In WCGS TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," Table 3.3.2-1, Footnotes (b) and (c) are added to the following SRs for the ESFAS Functions that meet the requirements of TSTF-493-A:

- SR 3.3.2.5, Perform a COT.
- SR 3.3.2.9, Perform a CHANNEL CALIBRATION.
- SR 3.3.2.12, Perform a COT.

Due to the addition of Footnotes (b) and (c) to TS Tables 3.3.1-1 and 3.3.2-1, the subsequent TS footnotes would be re-lettered as necessary.

For those instrument Functions not specifically addressed in TSTF-493-A, Attachment A, as described above, the TSTF specifies a Bases change. Instead of adding the Footnotes (b) and (c) to the TS, TSTF-493-A, Option A, requires the addition of a Bases statement to the TADOT (with setpoint verification), COT, and CHANNEL CALIBRATION SR Bases in the following NUREG-1431 Instrumentation TS:

- 3.3.5, "Loss of Power (LOP) Diesel Generator Start Instrumentation,"
- 3.3.6, "Containment Purge and Exhaust Isolation Instrumentation,"
- 3.3.7, "Control Room Emergency Filtration System (CREFS) Actuation Instrumentation,"
- 3.3.8, "Fuel Building Air Cleanup System (FBACS) Actuation Instrumentation," and
- 3.3.9, "Boron Dilution Protection System (BDPS)."

TSTF-493-A requires the following Bases statement to be added to the applicable SR Bases of the TS listed above:

> "There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology."

Consistent with the guidance of TSTF-493-A, the proposed change would add the Bases statement (above) to the SR Bases for the TADOT and CHANNEL CALIBRATION SRs of WCGS TS 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation." Note that there is no COT SR requirement associated with TS 3.3.5.

In addition, the TSTF-493-A Bases statement, described above, would be added to the SR Bases for the COT and CHANNEL CALIBRATION SRs of the following WCGS instrumentation TS:

- 3.3.6, "Containment Purge Isolation Instrumentation,"
- 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation," and
- 3.3.8, "Emergency Exhaust System (EES) Actuation Instrumentation."

Note that the TADOT SR for each of the WCGS TS listed above is modified by a Note that states "Verification of setpoint is not required." Therefore, consistent with the guidance of TSTF-493-A, the Bases statement, described above, is not proposed to be added to the Bases of the TADOT SRs of WCGS TS 3.3.6, TS 3.3.7, and TS 3.3.8. If a setpoint verification is not required, the TSTF-493-A Bases statement is not applicable to the TADOT SRs.

The WCGS TS do not include a corresponding TS to the NUREG-1431 TS 3.3.9, "Boron Dilution Protection System (BDPS)."

ALTERNATIVE SOURCE TERM (AST) IMPLEMENTATION

See Enclosure VI of this LAR for details associated with the implementation of the AST 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, "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 2.0 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 [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.

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) - Margin to Overfill (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 3) 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 3, and therefore the conclusions presented in Reference 3 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 4) 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 5) 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 Trip Setpoint of TS 3.3.1, Function 10, "Reactor Coolant Flow - Low" (identified in the current TS 3.3.1 Bases Table B 3.3.1-1) is:

> "≥ 89.9% of loop design flow (90,324 gpm)"

The proposed change would revise the Trip Setpoint to the NTSP for this RTS Function to:

> "90% of indicated loop flow"

The existing WCGS Trip Setpoint (≥ 89.9% of loop design flow (90,324 gpm) provided in the Bases of TS 3.3.1) is revised due to human factors (i.e. rounding the indicated flow value up to the next whole number of 90%) and also to be consistent with the assumptions of the new safety analysis methodology (the use of indicated loop flow, instead of design loop flow).

The RCS loss of flow events that credit the 90% of indicated loop RCS flow setpoint 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 ≥ 37.1×10^4 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 37.1×10^4 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 37.1×10^4 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 6) 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 3.7.10 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 7). 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 8) 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.

SETPOINT METHODOLOGY TRANSITION

The Westinghouse Setpoint Methodology, WCAP-17746-P, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station," (Enclosure II in this LAR) uses the NTSP instead of the Allowable Value for the LSSS specified in the TS. The specific calculations performed with the proposed NTSPs are provided in WCAP-17602-P, "Westinghouse Setpoint Calculations for the Wolf Creek Generating Station Control, Protection, and Indication Systems," (Enclosure IV of this LAR).

In the Westinghouse Setpoint Methodology, the NTSP is a predetermined setting for a protection channel chosen to ensure automatic actuation prior to a process variable reaching the analytical limit and thus ensuring that the applicable safety limits would not be exceeded. The NTSP accounts for uncertainties in setting the channel (e.g., calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the NTSP ensures that safety limits are not exceeded.

The following regulatory requirements are applicable to the instrument setpoint values specified in the TS.

Regulatory Guide 1.105, "Setpoints for Safety Related Instrumentation," states, in part:

> "...the LSSS establishes the threshold for protective system action to prevent acceptable limits being exceeded during design basis accidents. The LSSS therefore ensures that automatic protective action will correct the abnormal situation before a safety limit is exceeded."

10 CFR 50.36, "Technical specifications," specifies the required content of the TS and requires that the LSSS be included in the TS. 10 CFR 50.36(c)(1)(ii)(A) states, in part:

> "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded."

10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants." (GDC) Criterion 20, "Protection system functions," states:

> "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."

The proposed WCGS NTSPs, verified in accordance with the Westinghouse Setpoint Methodology (as described above), ensure that the required automatic reactor trips and safety feature actuations occur such that the safety limits are not exceeded. Therefore, the proposed NTSPs meet the requirements of Regulatory Guide 1.105, 10 CFR 50.36(c)(1)(ii)(A), and GDC 20 as stated above.

The Westinghouse Setpoint Methodology establishes a fixed-magnitude, two-sided (bi-directional) as-found tolerance (AFT) and as-left tolerance (ALT) about the NTSP, such that channel OPERABILITY is defined as the ability to be calibrated within the tolerance band. In the Westinghouse Setpoint Methodology, the ALT and the AFT are of the same magnitude, based on the same acceptance criterion. This method places relatively narrow limits on the allowable drift/deviation of field settings between successive surveillance intervals. Implementation of the Westinghouse Setpoint Methodology requires that each subsequent surveillance find the channel within the prescribed AFT about the NTSP, otherwise the instrument channel would be reset to a value that is within the ALT requirement around the NTSP as established by the Westinghouse Setpoint Methodology. Channels found to be not within the AFT are required to be evaluated. If the channel cannot be reset to within the ALT the channel would be declared inoperable. Based on this method, there is no need to specify an Allowable Value limit in the TS. This results in the Westinghouse Setpoint Methodology listing the NTSP as the LSSS TS limit. The Westinghouse Setpoint Methodology conservatively limits the allowed setpoint deviation from one surveillance to the next surveillance such that channels would be more likely to be readily identified as not performing as expected, such that the channel would be evaluated or remedial actions to restore the channel to OPERABLE status, would be taken, if required. Therefore, the implementation of the Westinghouse Setpoint Methodology provides adequate assurance that the affected instrument channels would continue to be maintained OPERABLE.

Detailed information describing the Westinghouse Setpoint Methodology as applied to the WCGS NTSPs is provided in Enclosures II and IV of this LAR.

The WCGS TS changes related to replacing the Allowable Values with NTSPs and the proposed WCGS NTSPs described in Section 2.0 (above) of this LAR (except as noted in Section 2.0 where the RCS flow setpoint (RTS Function 10) is changed from design to indicated flow) are based on the adoption of the Westinghouse Setpoint Methodology as described above and in Enclosures II and IV of this LAR.

IMPLEMENTATION OF TSTF-493-A

The Technical Analysis for this application is described in TSTF-493-A as referenced in the NRC Notice of Availability published in the Federal Register on May 11, 2010 (75 FR 26294). Plant specific information related to the Technical Analysis is described below to document that the content of TSTF-493-A, Revision 4, Option A, is applicable to WCGS.

Use of the Term Nominal Trip Setpoint

The term, Nominal Trip Setpoint (NTS), is the Westinghouse terminology for the setpoint value calculated by the plant specific Setpoint Methodology documented in WCAP-17746-P, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station" (Enclosure II of this LAR), which upon implementation of this LAR will be incorporated by

reference into the WCGS USAR. The term NTSP is the equivalent WCGS terminology for the setpoint value (the Westinghouse term of NTS), which is referred to in TSTF-493-A, Revision 4, and used in this LAR. The actual trip setpoint may be more conservative than the NTSP. In the case where the actual trip setpoint is more conservative than the NTSP, the ALT and AFT will be defined about the actual setting. The NTSP is the LSSS which is required to be in the TS by 10 CFR 50.36. 10 CFR 50.36(c)(1)(ii)(A) states, in part:

> "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions."

The NTSP specified in the WCGS TS is a predetermined setting for a protection channel chosen to ensure automatic actuation prior to the process variable reaching the analytical limit (AL) and thus ensuring that the SL would not be exceeded. In Enclosure II of this LAR, the safety analysis limit is the AL as referred to in TSTF-493-A, Revision 4. As such, the NTSP accounts for uncertainties in setting the channel (e.g., calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). The NTSP is the least conservative value (within an ALT) to which the channel must be reset at the conclusion of periodic testing to ensure that the AL will not be exceeded during an anticipated operational occurrence or accident before the next periodic surveillance or calibration. Relying solely on the NTSP to define OPERABILITY in the TS would be an overly restrictive requirement if it were applied as a limit for the "as-found" value of a protection channel setting during a surveillance. It is impossible to set a physical instrument channel to an exact value, so a calibration tolerance is established around the NTSP (the AFT and ALT bands).

The NTSPs are more conservative than the AL to account for applicable instrument measurement errors consistent with Enclosure II of this LAR. If during surveillance testing, the actual instrument setting is found less conservative than the AFT, the channel is evaluated for OPERABILITY consistent with proposed surveillance Footnote (b) which is discussed below. If the channel cannot be reset to within the ALT the channel is declared inoperable and actions must be taken consistent with the TS. An instrument adjustment is considered successful (i.e., the channel is OPERABLE) if the NTSP as-left instrument setting is within the ALT. For the Westinghouse Setpoint Methodology the field setting is the NTSP.

Addition of Channel Performance Surveillance Footnotes to TS Instrumentation Functions

The determination to include surveillance footnotes for specific Functions in the TS is based on these Functions being automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A). There are two surveillance footnotes added to the TS regarding the use of TS NTSPs for OPERABILITY determinations and for assessing channel performance. The Evaluation of Exclusion Criterion, section of this LAR (included below) discusses the principles applied to determine which Functions are to be annotated with the two surveillance footnotes.

The following table provides a comparison of the NUREG-1431, "Standard Technical Specifications Westinghouse Plants," (Reference 9) instrumentation Functions required to be annotated by TSTF-493-A (as identified in Attachment A of TSTF-493-A) and the WCGS TS. Plant specific differences between the guidance provided in TSTF-493-A, Attachment A and the WCGS TS are identified in the Table below.

TS Instrumentation Functions Required to be Annotated with SR Footnotes		
NUREG-1431 TS	**WCGS TS**	**Affected WCGS SRs**
TS 3.3.1, Reactor Trip System (RTS) Instrumentation Functions	**TS 3.3.1, Reactor Trip System (RTS) Instrumentation Functions**	
2.a Power Range Neutron Flux: High	2.a Power Range Neutron Flux: High	SR 3.3.1.7 SR 3.3.1.11
2.b Power Range Neutron Flux: Low	2.b Power Range Neutron Flux: Low	SR 3.3.1.8 SR 3.3.1.11
3.a Power Range Neutron Flux Rate: High Positive Rate	3.a Power Range Neutron Flux Rate: High Positive Rate	SR 3.3.1.7 SR 3.3.1.11
3.b Power Range Neutron Flux Rate: High Negative Rate	3.b Power Range Neutron Flux Rate: High Negative Rate	SR 3.3.1.7 SR 3.3.1.11
4. Intermediate Range Neutron Flux	4. Intermediate Range Neutron Flux	SR 3.3.1.8 SR 3.3.1.11
5. Source Range Neutron Flux	5. Source Range Neutron Flux	SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.11
6. Overtemperature ΔT	6. Overtemperature ΔT	SR 3.3.1.7 SR 3.3.1.10
7. Overpower ΔT	7. Overpower ΔT	SR 3.3.1.7 SR 3.3.1.10
8.a Pressurizer: Pressure Low	8.a Pressurizer: Pressure Low	SR 3.3.1.7 SR 3.3.1.10
8.b Pressurizer: Pressure High	8.b Pressurizer: Pressure High	SR 3.3.1.7 SR 3.3.1.10
9. Pressurizer Water level - High	9. Pressurizer Water level - High	SR 3.3.1.7 SR 3.3.1.10
10. Reactor Coolant Flow - Low	10. Reactor Coolant Flow - Low	SR 3.3.1.7 SR 3.3.1.10
12. Undervoltage RCPs	12. Undervoltage RCPs	SR 3.3.1.10
13. Underfrequency RCPs	13. Underfrequency RCPs	SR 3.3.1.10
14. Steam Generator (SG) Water Level – Low Low	14. Steam Generator (SG) Water Level Low – Low	SR 3.3.1.7 SR 3.3.1.10
15. SG Water Level – Low Coincident with Steam/Feedwater Flow Mismatch	15. Not Used. <u>Plant Specific Difference:</u> This Function is not included in the WCGS TS.	NA
16.a Turbine Trip: Low Fluid Oil Pressure	16.a Turbine Trip: Low Fluid Oil Pressure	SR 3.3.1.10

TS Instrumentation Functions Required to be Annotated with SR Footnotes		
NUREG-1431 TS	**WCGS TS**	**Affected WCGS SRs**
TS 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation Functions	**TS 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation Functions**	
1.c Safety Injection: Containment Pressure – High 1	1.c Safety Injection: Containment Pressure – High 1	SR 3.3.2.5 SR 3.3.2.9
1.d Safety Injection: Pressurizer Pressure – Low	1.d Safety Injection: Pressurizer Pressure – Low	SR 3.3.2.5 SR 3.3.2.9
1.e.(1) Safety Injection: Steam Line Pressure Low	1.e Safety Injection: Steam Line Pressure Low	SR 3.3.2.5 SR 3.3.2.9
1.e.(2) Safety Injection: High Differential Pressure Between Steam Lines	Plant Specific Difference: This Function is not included in the WCGS TS.	NA
1.f Safety Injection: High Steam Flow in Two Steam Lines Coincident with Tavg – Low Low	Plant Specific Difference: This Function is not included in the WCGS TS.	NA
1.g Safety Injection: High Steam Flow in Two Steam Lines Coincident with Steam Line Pressure - Low	Plant Specific Difference: This Function is not included in the WCGS TS.	NA
2.c Containment Spray: Containment Pressure High – 3 (High High)	2.c Containment Spray: Containment Pressure High – 3	SR 3.3.2.5 SR 3.3.2.9
2.d Containment Spray Containment Pressure High – 3 (Two Loop Plants)	Plant Specific Difference: This Function is not included in the WCGS TS.	NA
3.b.(3) Containment Phase B Isolation: Containment Pressure High – 3 (High High)	3.b.(3) Containment Phase B Isolation: Containment Pressure High – 3	SR 3.3.2.5 SR 3.3.2.9
4.c Steam Line Isolation: Containment Pressure – High 2	4.d Steam Line Isolation: Containment Pressure – High 2	SR 3.3.2.5 SR 3.3.2.9
4.d.(1) Steam Line Isolation: Steam Line Pressure Low	4.e(1) Steam Line Isolation: Steam Line Pressure Low	SR 3.3.2.5 SR 3.3.2.9
4.d.(2) Steam Line Isolation: Steam Negative Rate - High	4.e.(2) Steam Line Isolation: Negative Rate - High	SR 3.3.2.5 SR 3.3.2.9
4.e Steam Line Isolation: High Steam Flow in Two Steam Lines Coincident with Tavg – Low Low	Plant Specific Difference: This Function is not included in the WCGS TS.	NA
4.f Steam Line Isolation: High Steam Flow in Two Steam Lines Coincident with Steam Line Pressure – Low	Plant Specific Difference: This Function is not included in the WCGS TS.	NA

TS Instrumentation Functions Required to be Annotated with SR Footnotes		
NUREG-1431 TS	**WCGS TS**	**Affected WCGS SRs**
4.g Steam Line Isolation: High Steam Flow Coincident with Safety Injection Coincident with Tavg – Low Low	Plant Specific Difference: This Function is not included in the WCGS TS.	NA
4.h Steam Line Isolation: High High Steam Flow coincident with Safety Injection	Plant Specific Difference: This Function is not included in the WCGS TS.	NA
5.b Turbine Trip and Feedwater Isolation: SG Water Level – High High (P-14)	5.c Turbine Trip and Feedwater Isolation: SG Water Level – High High (P-14)	SR 3.3.2.5 SR 3.3.2.9
6.c Auxiliary Feedwater: SG Water Level – Low Low	6.d Auxiliary Feedwater: SG Water Level – Low Low	SR 3.3.2.5 SR 3.3.2.9
6.e Auxiliary Feedwater: Loss of Offsite Power	6.f Auxiliary Feedwater: Loss of Offsite Power Plant Specific Difference: Unlike NUREG-1431, this WCGS instrument Function does not have a setpoint specified in the TS which can be verified in accordance with the TSTF-493-A footnotes.	NA
6.f Auxiliary Feedwater: Undervoltage Reactor Coolant Pump	Plant Specific Difference: This Function is not included in the WCGS TS.	NA
6.g Auxiliary Feedwater: Trip of all Main Feedwater Pumps	6.g Auxiliary Feedwater: Trip of all Main Feedwater Pumps Plant Specific Difference: Unlike NUREG-1431, this WCGS instrument Function does not have a setpoint specified in the TS which can be verified in accordance with the TSTF-493-A footnotes.	NA
6.h Auxiliary Feedwater: Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	6.h Auxiliary Feedwater: Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	SR 3.3.2.9 SR 3.3.2.12
7.b Automatic Switchover to Containment Sump: Refueling Water Storage Tank (RWST) Level – Low Low Coincident with Safety Injection	7.b Automatic Switchover to Containment Sump: Refueling Water Storage Tank (RWST) Level – Low Low Coincident with Safety Injection	SR 3.3.2.5 SR 3.3.2.9
7.c Automatic Switchover to Containment Sump: RWST Level – Low Low Coincident with Safety Injection and Coincident with Containment Sump Level – High	Plant Specific Difference: This Function is not included in the WCGS TS.	NA

The following footnotes would be added to the SRs for the TS instrumentation Functions identified above:

"(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service."

"(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and the as-left tolerances is specified in WCAP-17746-P."

Setpoint calculations establish an NTSP based on the AL of the safety analysis to ensure that trips or protective actions will occur prior to exceeding the process parameter value assumed by the safety analysis calculations. These setpoint calculations also calculate an allowed limit of expected change (i.e., the AFT) between performances of the surveillance test for assessing the value of the setpoint setting. The least conservative as-found instrument setting value that a channel can have during calibration without requiring performing a TS remedial action is defined by the AFT. Discovering an instrument setting to be less conservative than allowed by the AFT indicates that there may not be sufficient margin between the setting and the AL. TS CHANNEL CALIBRATIONs, COTs, and TADOTs (with setpoint verification) are performed to verify channels are operating within the assumptions of the setpoint methodology calculated NTSP and that channel settings have not exceeded the AFT associated with the NTSP. When the measured as-found setpoint is non-conservative with respect to the NTSP AFT, the channel must be evaluated in accordance with the proposed surveillance Footnote (b). If the channel cannot be reset to within the ALT (surveillance Footnote (c)), the channel is declared inoperable and the actions identified in the TS must be taken.

Surveillance Footnote (b) requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its AFT. Evaluation of channel performance will verify that the channel will continue to perform in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service.

Verifying that a trip setting is within the AFT of the NTSP when a surveillance test is performed confirms OPERABILITY for that surveillance. Although the channel was OPERABLE during the previous surveillance interval, if it is discovered that channel performance is outside the performance predicted by the plant setpoint calculations for the test interval (i.e., outside the AFT), then the design basis for the channel may not be met, and proper operation of the channel for a future demand cannot be assured. Surveillance Footnote (b) formalizes the establishment of the appropriate AFT for each channel. This AFT is applied about the NTSP or about any other more conservative setpoint. The AFT ensures that channel operation is consistent with the assumptions or design inputs used in the setpoint calculations and establishes a high confidence of acceptable channel performance in the future. Because the AFT allows for both conservative and non-conservative deviation from the NTSP, changes in channel performance that are conservative with respect to the NTSP will also be detected and evaluated for possible effects on expected performance.

To implement surveillance Footnote (c), the ALT for some instrumentation Function channels is established to ensure that realistic values are used that do not mask instrument performance. Setpoint calculations assume that the instrument setpoint is left at the NTSP within a specific ALT (e.g., 25 psig ± 2 psig). A tolerance band is necessary because it is not possible to read and adjust a setting to an absolute value due to the readability and/or accuracy of the test instruments or the ability to adjust potentiometers. The ALT is normally as small as possible considering the tools and the objective to meet an as low as reasonably achievable calibration setting of the instruments. The ALT is determined in the setpoint calculation. Failure to set the actual plant trip setpoint to the NTSP (or more conservative than the NTSP), and within the ALT, would invalidate the assumptions in the setpoint calculation because any subsequent instrument drift would not start from the expected as-left setpoint.

Evaluation of Exclusion Criteria

Exclusion criteria are used to determine which Functions do not need to receive the proposed footnotes, as discussed in TSTF-493-A, Revision 4. Instruments are excluded from the additional requirements when their functional purpose can be described as (1) a manual actuation circuit, (2) an automatic actuation logic circuit, or (3) an instrument function that derives input from contacts which have no associated sensor or adjustable device. Many permissives or interlocks are excluded if they derive input from a sensor or adjustable device that is tested as part of another TS function. The list of affected Functions identified in the table below was developed on the principle that all Functions in the affected TS are included unless one or more of the exclusion criterion apply. If the excluded Functions differ from the list of excluded Functions in TSTF-493-A, Revision 4, a justification for that deviation is provided.

The Table below compares the excluded instrumentation Functions from NUREG-1431 identified in Attachment A of TSTF-493-A to the list of WCGS instrumentation Functions identified for exclusion. Plant specific differences are identified and justified in the Table below.

TS Instrumentation Functions Excluded From Having SR Footnotes Added	
NUREG-1431 TS	**WCGS TS**
TS 3.3.1, Reactor Trip System (RTS) Instrumentation Functions	**TS 3.3.1, Reactor Trip System (RTS) Instrument Functions**
1. Manual Reactor Trip (Manual Actuation excluded from footnotes)	1. Manual Reactor Trip (Manual Actuation excluded from footnotes)
11. Reactor Coolant Pump (RCP) Breaker Position (Mechanical component excluded from footnotes)	11. Not Used. Plant Specific Difference: This Function is not included in the WCGS TS.
16.b Turbine Trip: Turbine Stop Valve Closure (Mechanical component excluded from footnotes)	16.b Turbine Trip: Turbine Stop Valve Closure (Mechanical component excluded from footnotes)
17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS) (Automatic actuation logic circuit excluded from footnotes)	17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS) (Automatic actuation logic circuit excluded from footnotes)

TS Instrumentation Functions Excluded From Having SR Footnotes Added	
NUREG-1431 TS	**WCGS TS**
18. Reactor Trip System Interlocks (Permissive or interlock excluded from footnotes if they derive input from a sensor or adjustable device that is tested as part of another TS function)	18. Reactor Trip System Interlocks (Permissive or interlock excluded from footnotes) WCNOC confirms the interlocks derive input from a sensor or adjustable device that is tested as part of another TS function
19. Reactor Trip Breakers (RTBs) (Mechanical component excluded from footnotes)	19. Reactor Trip Breakers (RTBs) (Mechanical component excluded from footnotes)
20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms (Mechanical component excluded from footnotes)	20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms (Mechanical component excluded from footnotes)
21. Automatic Trip Logic (Automatic actuation logic circuit excluded from footnotes)	21. Automatic Trip Logic (Automatic actuation logic circuit excluded from footnotes)
TS 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation Functions	**TS 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation Functions**
1.a Safety Injection: Manual Initiation (Manual actuation excluded from footnotes)	1.a Safety Injection: Manual Initiation (Manual actuation excluded from footnotes)
1.b Safety Injection: Automatic Actuation Logic and Actuation Relays (Automatic actuation logic circuit excluded from footnotes)	1.b Safety Injection: Automatic Actuation Logic and Actuation Relays (Automatic actuation logic circuit excluded from footnotes)
2.a Containment Spray: Manual Initiation - (Manual actuation excluded from footnotes)	2.a Containment Spray: Manual Initiation - (Manual actuation excluded from footnotes)
2.b Containment Spray: Automatic Actuation Logic and Actuation Relays (Automatic actuation logic circuit excluded from footnotes)	2.b Containment Spray: Automatic Actuation Logic and Actuation Relays (Automatic actuation logic circuit excluded from footnotes)
3.a.(1) Containment Isolation: Phase A Isolation Manual Initiation (Manual actuation excluded from footnotes)	3.a.(1) Containment Isolation: Phase A Isolation Manual Initiation (Manual actuation excluded from footnotes)
3.a.(2) Containment Isolation: Phase A Isolation Automatic Actuation Logic and Actuation Relays (Automatic actuation logic circuit excluded from footnotes)	3.a.(2) Containment Isolation: Phase A Isolation Automatic Actuation Logic and Actuation Relays (Automatic actuation logic circuit excluded from footnotes)
3.a.(3) Containment Isolation: Phase A Isolation Safety Injection (Automatic actuation logic circuit excluded from footnotes)	3.a.(3) Containment Isolation: Phase A Isolation Safety Injection (Automatic actuation logic circuit excluded from footnotes)
3.b.(1) Containment Isolation: Phase B Isolation Manual Initiation (Manual actuation excluded from footnotes)	3.b.(1) Containment Isolation: Phase B Isolation Manual Initiation (Manual actuation excluded from footnotes)
3.b.(2) Containment Isolation: Phase B Isolation Automatic Actuation Logic and Actuation Relays (Automatic actuation logic circuit excluded from footnotes)	3.b.(2) Containment Isolation: Phase B Isolation Automatic Actuation Logic and Actuation Relays (Automatic actuation logic circuit excluded from footnotes)
4.a Steam Line Isolation: Manual Initiation (Manual actuation excluded from footnotes)	4.a Steam Line Isolation: Manual Initiation (Manual actuation excluded from footnotes)

TS Instrumentation Functions Excluded From Having SR Footnotes Added	
NUREG-1431 TS	**WCGS TS**
4.b Steam Line Isolation: Automatic Actuation Logic and Actuation Relays (Automatic actuation logic circuit excluded from footnotes)	4.b Steam Line Isolation: Automatic Actuation Logic and Actuation Relays (SSPS) (Automatic actuation logic circuit excluded from footnotes)
NA	4.c Steam Line Isolation: Automatic Actuation Logic (MSFIS) (Automatic actuation logic circuit excluded from footnotes) <u>Plant Specific Difference:</u> The WCGS design includes an additional automatic actuation logic circuit associated with Steam Line Isolation (Function 4.c) that is not part of the TS in NUREG-1431.
4.g Steam Line Isolation: High Steam Flow Coincident with Safety Injection (Automatic actuation logic circuit excluded from footnotes)	<u>Plant Specific Difference:</u> This Function is not included in the WCGS TS.
4.h Steam Line Isolation: High High Steam Flow Coincident with Safety Injection (Automatic actuation logic circuit excluded from footnotes)	<u>Plant Specific Difference:</u> This Function is not included in the WCGS TS.
5.a Turbine Trip and Feedwater Isolation: Automatic Actuation Logic and Actuation Relays (Automatic actuation logic circuit excluded from footnotes)	5.a Turbine Trip and Feedwater Isolation: Automatic Actuation Logic and Actuation Relays (SSPS) (Automatic actuation logic circuit excluded from footnotes)
NA	5.b Turbine Trip and Feedwater Isolation: Automatic Actuation Logic (MSFIS) (Automatic actuation logic circuit excluded from footnotes) <u>Plant Specific Difference:</u> The WCGS design includes an additional automatic actuation logic circuit associated with Turbine Trip and Feedwater Isolation (Function 5.b) that is not part of the TS in NUREG-1431.
5.c Turbine Trip and Feedwater Isolation: Safety Injection (Automatic actuation logic circuit excluded from footnotes)	5.d Turbine Trip and Feedwater Isolation: Safety Injection (Automatic actuation logic circuit excluded from footnotes)
NA	6.a Auxiliary Feedwater: Manual Initiation (Manual actuation excluded from footnotes) <u>Plant Specific Difference:</u> Auxiliary Feedwater Manual Initiation is not included in the NUREG-1431 TS.
6.a Auxiliary Feedwater: Automatic Actuation Logic and Actuation Relays (Solid State Protection System) (Automatic actuation logic circuit excluded from footnotes)	6.b Auxiliary Feedwater: Automatic Actuation Logic and Actuation Relays (Solid State Protection System) (Automatic actuation logic circuit excluded from footnotes)

TS Instrumentation Functions Excluded From Having SR Footnotes Added	
NUREG-1431 TS	**WCGS TS**
6.b Auxiliary Feedwater: Automatic Actuation Logic and Actuation Relays (Balance of Plant ESFAS) (Automatic actuation logic circuit excluded from footnotes)	6.c Auxiliary Feedwater: Automatic Actuation Logic and Actuation Relays (Balance of Plant ESFAS) (Automatic actuation logic circuit excluded from footnotes)
6.d Auxiliary Feedwater: Safety Injection (Automatic actuation logic circuit excluded from footnotes)	6.e Auxiliary Feedwater: Safety Injection (Automatic actuation logic circuit excluded from footnotes)
NA	6.f Auxiliary Feedwater: Loss of Offsite Power <u>Plant Specific Difference:</u> Unlike NUREG-1431, this WCGS instrumentation Function has no setpoint specified in the current TS and Bases which can be verified in accordance with the TSTF-493-A footnotes. The setpoint methodology specified in the second TSTF-493-A footnote is not applicable to this instrumentation Function. Therefore, no footnotes are proposed to be added to this WCGS instrumentation Function.
NA	6.g Auxiliary Feedwater: Trip of all Main Feedwater Pumps. <u>Plant Specific Difference:</u> Unlike NUREG-1431, this WCGS instrumentation Function has no setpoint specified in the current TS and Bases which can be verified in accordance with the TSTF-493-A footnotes. The setpoint methodology specified in the second TSTF-493-A footnote is not applicable to this instrumentation Function. Therefore, no footnotes are proposed to be added to this WCGS instrumentation Function.
7.a Automatic Switchover to Containment Sump: Automatic Actuation Logic and Actuation Relays (Automatic actuation logic circuit excluded from footnotes)	7.a Automatic Switchover to Containment Sump: Automatic Actuation Logic and Actuation Relays (Automatic actuation logic circuit excluded from footnotes)
7.b Automatic Switchover to Containment Sump: Refueling Water Storage Tank (RWST) Level – Low Low Coincident with Safety Injection (Automatic actuation logic circuit excluded from footnotes)	7.b Automatic Switchover to Containment Sump: Refueling Water Storage Tank (RWST) Level – Low Low Coincident with Safety Injection (Automatic actuation logic circuit excluded from footnotes)
7.c Automatic Switchover to Containment Sump: Refueling Water Storage Tank (RWST) Level – Low Low Coincident with Safety Injection (Automatic actuation logic circuit excluded from footnotes) and Coincident with Containment Sump Level - High	<u>Plant Specific Difference:</u> This Function is not included in the WCGS TS.

TS Instrumentation Functions Excluded From Having SR Footnotes Added	
NUREG-1431 TS	**WCGS TS**
8. ESFAS Interlocks (Permissive or interlock excluded from footnotes if they derive input from a sensor or adjustable device that is tested as part of another TS function)	8. ESFAS Interlocks (Permissive or interlock excluded from footnotes) WCNOC confirms the interlocks derive input from a sensor or adjustable device that is tested as part of another TS function

Setpoint Values

WCNOC is proposing changes to the instrument setting limits in the TS. The changes to the existing TS setpoints are described in detail in Section 2.0 of this LAR in the "Setpoint Methodology Transition" discussion. Additional justification for the setpoint changes is provided in this section of the LAR (3.0), and also in the "Setpoint Methodology Transition" discussion.

FULL IMPLEMENTATION OF ALTERNATIVE SOURCE TERM

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

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

CORE DESIGN AND SAFETY ANALYSIS METHODOLOGY TRANSITION

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, "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.

SETPOINT METHODOLOGY TRANSITION

10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," 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.

10 CFR 50.36, "Technical specifications," specifies the required content of the TS and requires that LSSS be included in the TS. Specifically, 10 CFR 50.36(c)(1)(ii)(A) states, in part:

> "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded."

Regulatory Guide (RG) 1.105, "Setpoints For Safety-Related Instrumentation," describes a method acceptable to the NRC staff for complying with the NRC's regulations for ensuring that setpoints for safety related instrumentation are initially within and remain within the technical specification limits.

IMPLEMENTATION OF TSTF-493-A

WCNOC has reviewed the NRC staff's model SE published as part of the Notice of Availability and concluded that the regulatory evaluation section is applicable to WCGS.

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 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.

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) 3.3.1, "Reactor Trip System (RTS) Instrumentation," TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start 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 Nuclear Regulatory Commission (NRC) approved analysis methodologies. As part of the transition to the standard Westinghouse developed methodologies, an instrument uncertainty analysis was performed based on the current Westinghouse Setpoint Methodology. This amendment request includes the adoption of Option A of Technical Specification Task Force

(TSTF) TSTF-493-A, Revision 4, "Clarify Application of Setpoint Methodology for LSSS Functions."

In addition, the proposed amendment request revises the TS definitions of DOSE EQUIVALENT I-131, and DOSE EQUIVALENT XE-133, and Specification 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring 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 Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0. In conjunction with the full scope implementation of AST, the proposed amendment request includes changes to adopt TSTF-51-A, Revision 2, "Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations." The adoption of TSTF-51-A results in changes to TS 3.3.6, "Containment Purge Isolation Instrumentation," TS 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation," TS 3.3.8, "Emergency Exhaust System (EES) Actuation Instrumentation," and TS 3.7.10, "Control Room Emergency Ventilation System (CREVS)," TS 3.7.11, "Control Room Air Conditioning System (CRACS)," TS 3.7.13, "Emergency Exhaust System (EES)," and TS 3.9.4, "Containment Penetrations."

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 proposed changes associated with the implementation of Technical Specification Task Force (TSTF)-493-A adds test requirements to TS instrumentation functions related to those variables that have a significant safety function to ensure that instruments will function as required to initiate protective systems or actuate mitigating systems as assumed in the safety analysis. The proposed changes do not impact the condition or performance of any plant structure, system or component. The new core design, non-loss-of-coolant-accident (non-LOCA) and Post-LOCA Subcriticality and Cooling analyses and the proposed Nominal Trip Setpoints (NTSPs) 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, including the setpoint methodology 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 TS changes associated with the implementation of TSTF-493-A will provide additional assurance that the instrumentation setpoints are maintained consistent with the setpoint methodology to ensure the required automatic trips and safety feature actuations occur such that the safety limits are not exceeded. The requested TS changes, including those changes proposed to conform to the new methodologies and TSTF-493-A 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 (including those changes resulting from the adoption of TSTF-51-A) 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

The proposed change involves a physical alteration of the plant, i.e., a change in instrument setpoint. 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 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 after the AST is implemented, therefore, no new failure modes are created by this proposed change. The AST change does not involve changing any accident initiators.

For the fuel handling accident, the adoption of TSTF-51-A permits the elimination of the TS requirements for certain Engineered Safety Feature (ESF) systems to be OPERABLE after sufficient radioactive decay. However, after sufficient radioactive decay, no credit is taken for these ESF systems to meet the applicable regulatory dose limits in the event of a fuel handling accident. Therefore, no structures, systems and components important to safety are adversely affected by the proposed change. The proposed change resulting from the adoption of TSTF-51-A does not involve changing any accident initiators.

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 changes and implementation of TSTF-493-A 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.

For the fuel handling accident, the adoption of TSTF-51-A allows the elimination of the TS requirements for certain ESF systems to be OPERABLE, after sufficient radioactive decay. However, after sufficient radioactive decay, no credit is taken for these ESF systems to meet the applicable regulatory dose limits in the event of a fuel handling accident. Therefore, no structures, systems and components important to safety are adversely affected by the proposed change. With the proposed changes, the requirements of the TS will reflect that after sufficient radioactive decay, the water level and decay time inputs will be the primary success path for mitigating a fuel handling accident. Thus, the TS will continue to provide adequate assurance of safe operation during fuel handling. As such, there is no significant reduction in a margin of safety.

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 Commission's regulations, and 3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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. Regulatory Guide 1.105, Revision 3, "Setpoints For Safety-Related Instrumentation," December 1999.

3. 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 (TAC No. ME5186) (Proprietary)," February 17, 2012.

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

5. 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.

6. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," January 1999.

7. NRC letter from J. Donohew to O.L. Maynard, "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 02018190.

8. NRC letter from J. R. Jolicoeur to TSTF, "Implementation Of Travelers TSTF-363, Revision 0, "Revise Topical Report References ITS 5.6.5, COLR [CORE OPERATING LIMITS REPORT]," TSTF-408, Revision 1, "Relocation of LTOP [LOW TEMPERATURE OVERPRESSURE PROTECTION] Enable Temperature and PORV [POWER-OPERATED RELIEF VALVE] Lift Setting to the PTLR [PRESSURE-TEMPERATURE LIMITS REPORT], AND TSTF-419, Revision 0, "Revise PTLR Definition and References in ISTS [IMPROVED STANDARD TECHNICAL SPECIFICATION] 5.6.6, RCS [REACTOR COOLANT SYSTEM] PTLR," August 4, 2011, ADAMS Accession no. ML 110660285.

9. NUREG-1431, Revision 4, "Standard Technical Specifications Westinghouse Plants," April 2012.

Proposed Technical Specification Changes (Mark-ups)

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.

2.1.2 RCS Pressure SL

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

2.2 SL Violations

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

2.2.2 If SL 2.1.2 is violated:

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

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

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

NOMINAL TRIP SETPOINT

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)~~ (a)	2	C	SR 3.3.1.14	NA
2. Power Range Neutron Flux					109%
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	~~≤ 112.3%~~ RTP (b)(c)
	d				25%
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					(b)(c) 4%
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% (b)(c)
	d e				
4. Intermediate Range Neutron Flux	1~~(e)~~, 2~~(d)~~	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	~~≤ 35.3%~~ RTP 25%
	f				
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 10^5 (b)(c)
	a				
	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.

(a) (d) (e) (f)

INSERT 1

INSERT 1

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and the as-left tolerances is specified in WCAP-17746-P.

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

NOMINAL TRIP SETPOINT

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	~~ALLOWABLE VALUE~~ ~~(a)~~
6. Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 1 (Page 3.3-19)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 2 (Page 3.3-20)
8. Pressurizer Pressure					
a. Low	1(g)	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	~~≥ 1930~~ psig 1940
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	~~≤ 2395~~ psig 2385
9. Pressurizer Water Level - High	1(g)	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	~~≤ 93.9%~~ of instrument span 92%
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)~~ 90% of indicated loop 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 10578

(b)(c)

(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.~~

INSERT 1

INSERT 1

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and the as-left tolerances is specified in WCAP-17746-P.

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	~~ALLOWABLE VALUE (a)~~ NOMINAL TRIP SETPOINT
13. Underfrequency RCPs	1(g)	2/bus	M	SR 3.3.1.9 SR 3.3.1.10 (b)(c) SR 3.3.1.16	~~≥ 57.1~~ 57.2 Hz
14. Steam Generator (SG) Water Level Low-Low (l)	1,2	4 per gen	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 (b)(c) SR 3.3.1.16	~~≥ 22.3%~~ 23.5% of Narrow Range Instrument Span
15. Not Used.					
16. Turbine Trip					
a. Low Fluid Oil Pressure	1(j)	3	O	SR 3.3.1.10 (b)(c) SR 3.3.1.15	~~≥ 534.20~~ 590.0 psig
b. Turbine Stop Valve Closure	1(j)	4	P	SR 3.3.1.10 SR 3.3.1.15	≥ 1% open
17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	Q	SR 3.3.1.14	NA
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2(e) f	2	S	SR 3.3.1.11 SR 3.3.1.13	~~≥ 6E-11 amp~~ 1.0E-10 amps
b. Low Power Reactor Trips Block, P-7	1	1 per train	T	SR 3.3.1.5	NA
c. Power Range Neutron Flux, P-8	1	4	T	SR 3.3.1.11 SR 3.3.1.13	~~≤ 51.3%~~ 48% RTP

(continued)

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

f

INSERT 1

INSERT 1

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and the as-left tolerances is specified in WCAP-17746-P.

Table 3.3.1-1 (page 4 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	~~ALLOWABLE VALUE (a)~~ NOMINAL TRIP SETPOINT
18. (continued)					
d. Power Range Neutron Flux, P-9	1	4	T	SR 3.3.1.11 SR 3.3.1.13	~~≤ 53.3%~~ 50% RTP
e. Power Range Neutron Flux, P-10	1,2	4	S	SR 3.3.1.11 SR 3.3.1.13	~~≥ 6.7%~~ 10% RTP ~~and ≤ 13.3% RTP~~
f. Turbine Impulse Pressure, P-13	1	2	T	SR 3.3.1.10 SR 3.3.1.13	~~≤ 12.4%~~ 10% turbine power
19. Reactor Trip Breakers (RTB) (k)	a 1,2	2 trains	R	SR 3.3.1.4	NA
	3(~~b~~), 4(~~b~~), 5(~~b~~)	2 trains	C	SR 3.3.1.4	NA
20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms (k)	a 1,2	1 each per RTB	U	SR 3.3.1.4	NA
	3(~~b~~), 4(~~b~~), 5(~~b~~)	1 each per RTB	C	SR 3.3.1.4	NA
21. Automatic Trip Logic	a 1,2	2 trains	Q	SR 3.3.1.5	NA
	3(~~b~~), 4(~~b~~), 5(~~b~~)	2 trains	C	SR 3.3.1.5	NA

~~(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.
a (k) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

~~The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.3% of ΔT span.~~

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \le \Delta T_O \left\{ K_1 - K_2 \frac{(1+\tau_4 s)}{(1+\tau_5 s)}\left[T\left(\frac{1}{(1+\tau_6 s)}\right) - T' \right] + K_3 (P - P') - f_1(\Delta I) \right\}$$

loop

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T′ is the nominal T_{avg} at RTP, ≤ *.

(T_{ref} from Rod Control)

P is the measured pressurizer pressure, psig.
P′ is the nominal RCS operating pressure ≥ * psig.

K_1 = *	K_2 = * /°F	K_3 = * /psig
τ_1 = * sec	τ_2 = * sec	τ_3 = * sec
τ_4 = * sec	τ_5 = * sec	τ_6 = * sec

$f_1(\Delta I)$ =	* { * % + (q_t - q_b)}	when q_t - q_b < * % RTP
	0% of RTP	when * % RTP ≤ q_t - q_b ≤ * % RTP
	* {(q_t - q_b) - * % }	when q_t - q_b > * % RTP

where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and q_t + q_b is the total THERMAL POWER in percent RTP.

The values denoted with * are specified in the COLR.

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

Note 2: Overpower ΔT

~~The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 2.6% of ΔT span.~~

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \le \Delta T_O \left\{ K_4 - K_5 \frac{(\tau_7 s)}{(1+\tau_7 s)}\left(\frac{1}{1+\tau_6 s}\right) T - K_6 \left[T \frac{1}{(1+\tau_6 s)} - T'' \right] - f_2(\Delta I) \right\}$$

loop

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T″ is the ~~indicated~~ T_{avg} at RTP ~~(Calibration temperature for ΔT instrumentation)~~, ≤ * °F.

K_4 = *
nominal

K_5 = * /°F for increasing T_{avg}
* /°F for decreasing T_{avg}

K_6 = * /°F when T > T″
* /°F when T ≤ T″

τ_1 = * sec
τ_2 = * sec
(T_{ref} from Rod Control)
τ_3 = * sec
τ_6 = * sec
τ_7 = * sec

$f_2(\Delta I)$ = *

The values denoted with * are specified in the COLR.

Table 3.3.2-1 (page 1 of 5)
Engineered Safety Feature Actuation System Instrumentation

NOMINAL TRIP SETPOINT

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	~~ALLOWABLE VALUE (a)~~
1.	Safety Injection					
a.	Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA
b.	Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
c.	Containment Pressure - High 1	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ ~~4.5~~ psig
d.	Pressurizer Pressure - Low	1,2,3~~(b)~~	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ ~~1820~~ psig
e.	Steam Line Pressure Low	1,2,3~~(b)~~	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ ~~674~~ psig~~(e)~~
2.	Containment Spray					
a.	Manual Initiation	1,2,3,4	2 per train, 2 trains	B	SR 3.3.2.8	NA
b.	Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
c.	Containment Pressure High - 3	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ ~~28.3~~ psig

3.5
1830
(b)(c)
a
615
d
(b)(c)
27.0
(b)(c)

(continued)

a
~~(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.~~
(b) Above the P-11 (Pressurizer Pressure) interlock and below P-11 unless the Function is blocked.
(c) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.
INSERT 1
d

INSERT 1

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and the as-left tolerances is specified in WCAP-17746-P.

Table 3.3.2-1 (page 2 of 5)
Engineered Safety Feature Actuation System Instrumentation

NOMINAL TRIP SETPOINT

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	~~ALLOWABLE VALUE (a)~~
3. Containment Isolation					
a. Phase A Isolation					
(1) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
b. Phase B Isolation					
(1) Manual Initiation	1,2,3,4	2 per train, 2 trains	B	SR 3.3.2.8	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
(3) Containment Pressure - High 3	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ ~~28.3~~ psig
4. Steam Line Isolation					
a. Manual Initiation	1,2[i], 3[i]	2	F	SR 3.3.2.8	NA
b. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2[i], 3[i]	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
c. Automatic Actuation Logic (MSFIS)	1,2[l], 3[l]	2 trains	G	SR 3.3.2.6	NA
d. Containment Pressure - High 2	1,2[i], 3[i]	3	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ ~~18.3~~ psig

27.0

(b)(c)

17.0

(b)(c)

INSERT 1

(continued)

~~(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.~~
(i) Except when all MSIVs are closed and de-activated; and all MSIV bypass valves are closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves.
(l) Except when all MSIVs are closed and de-activated.

INSERT 1

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and the as-left tolerances is specified in WCAP-17746-P.

Table 3.3.2-1 (page 3 of 5)
Engineered Safety Feature Actuation System Instrumentation

NOMINAL TRIP SETPOINT

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	~~ALLOWABLE VALUE~~ ~~(a)~~
4. Steam Line Isolation (continued)					
e. Steam Line Pressure					
(1) Low	1,2(i),3(~~b~~)(i)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ ~~671~~ 615 psig(~~c~~) d
(2) Negative Rate - High	3(g)(i)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ ~~125~~ 100(h) psi
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2(j),3(j)	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
b. Automatic Actuation Logic (MSFIS)	1,2(k),3(k)	2 trains	G	SR 3.3.2.6	NA
c. SG Water Level -High High (P-14)	1,2(j)	4 per SG	I	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ ~~79.7%~~ 78% of Narrow Range Instrument Span
d. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				

(b)(c)

(continued)

a

~~(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.~~
(~~b~~) Above the P-11 (Pressurizer Pressure) Interlock and below P-11 unless the Function is blocked.
(~~c~~) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.
(~~g~~) Below the P-11 (Pressurizer Pressure) Interlock; however, may be blocked below P-11 when safety injection on low steam line pressure is not blocked.
(h) Time constant utilized in the rate/lag controller is ≥ 50 seconds.
(i) Except when all MSIVs are closed and de-activated; and all MSIV bypass valves are closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves.
(j) Except when all MFIVs are closed and de-activated; and all MFRVs are closed and de-activated or closed and isolated by a closed manual valve; and all MFRV bypass valves are closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves.
(k) Except when all MFIVs are closed and de-activated.

d

INSERT 1

INSERT 1

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and the as-left tolerances is specified in WCAP-17746-P.

Table 3.3.2-1 (page 4 of 5)
Engineered Safety Feature Actuation System Instrumentation

NOMINAL TRIP SETPOINT

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	~~ALLOWABLE VALUE (a)~~
6. Auxiliary Feedwater					
a. Manual Initiation	1,2,3	1 per pump	O	SR 3.3.2.8	NA
b. Automatic Actuation Logic and Actuation Relays (Solid State Protection System)	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
c. Automatic Actuation Logic and Actuation Relays (Balance of Plant ESFAS)	1,2,3	2 trains	N	SR 3.3.2.3	NA
d. SG Water Level Low - Low	1,2,3	4 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ ~~22.3%~~ of Narrow Range Instrument Span
e. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
f. Loss of Offsite Power	1,2,3	2 trains	P	SR 3.3.2.7 SR 3.3.2.10	NA
g. Trip of all Main Feedwater Pumps	1	2 per pump	J	SR 3.3.2.8	NA
h. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	1,2,3	3	M	SR 3.3.2.1 SR 3.3.2.9 SR 3.3.2.10 SR 3.3.2.12	≥ ~~20.53~~ psia

(b)(c)

23.5%

21.6

INSERT 1

(continued)

~~(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.~~

INSERT 1

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and the as-left tolerances is specified in WCAP-17746-P.

Table 3.3.2-1 (page 5 of 5)
Engineered Safety Feature Actuation System Instrumentation

NOMINAL TRIP SETPOINT

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	~~ALLOWABLE VALUE(a)~~
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
b. Refueling Water Storage Tank (RWST) Level - Low Low	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ ~~35.5%~~ of instrument span
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
8. ESFAS Interlocks					
a. Reactor Trip, P-4(m)	1,2,3	2 per train, 2 trains	F	SR 3.3.2.11	NA
b. Pressurizer Pressure, P-11	1,2,3	3	L	SR 3.3.2.5 SR 3.3.2.9	≤ ~~1979~~ psig

36.0%

(b)(c)

1970

INSERT 1

~~(a) The Allowable Value defines the Limiting Safety System Settings. See the Bases for the Trip Setpoints.~~

(m) The functions of the Reactor Trip, P-4 interlock required to meet the LCO are:
- Trips the main turbine – MODES 1 and 2
- Isolates MFW with coincident low T_{avg} – MODES 1 and 2
- Allows manual block of the automatic reactuation of SI after a manual reset of SI – MODES 1, 2, and 3
- Prevents opening of MFIVs if closed on SI or SG Water Level – High High – MODES 1, 2, and 3

INSERT 1

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and the as-left tolerances is specified in WCAP-17746-P.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.5.1	Not Used.	
SR 3.3.5.2	----------NOTE---------- Verification of time delays is not required. ---------- Perform TADOT.	31 days
SR 3.3.5.3	Perform CHANNEL CALIBRATION ~~with nominal Trip Setpoint and Allowable Value~~ as follows: a. ~~Loss of voltage Allowable Value ≥ 82.5V, 120V bus with a time delay of 1.0 + 0.2, -0.5 sec.~~ Loss of voltage ~~nominal~~ Nominal Trip Setpoint 83V, 120V bus with a time delay of 1.0 sec. b. ~~Degraded voltage Allowable Value ≥ 105.9V, 120V bus with a time delay of 119 ± 11.6 sec.~~ Degraded voltage ~~nominal~~ Trip Setpoint ~~106.9~~ 108.7V, 120V bus with a time delay of 119 sec.	18 months
SR 3.3.5.4	Verify LOP DG Start ESF RESPONSE TIMES are within limits.	18 months on a STAGGERED TEST BASIS

3.4 REACTOR COOLANT SYSTEM (RCS)

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

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

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

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

c. RCS total flow rate ≥ ~~37.1×10^4~~ 361,200 gpm and greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 1.

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

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

b. THERMAL POWER step > 10% RTP.

ACTIONS

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.1.2	Verify RCS average temperature is less than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ ~~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

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

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

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

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

1. ~~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)

5.6 Reporting Requirements 2

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

~~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."~~

4 ~~7.~~ WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code."

5 ~~8.~~ WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores."

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

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

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

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

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

(continued)

Revised Technical Specification Pages

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

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

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

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

2.1.2 RCS Pressure SL

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

2.2 SL Violations

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

2.2.2 If SL 2.1.2 is violated:

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

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

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	NOMINAL TRIP SETPOINT
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA
	3[a], 4[a], 5[a]	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[b][c] SR 3.3.1.11[b][c] SR 3.3.1.16	109% RTP
b. Low	1[d],2	4	E	SR 3.3.1.1 SR 3.3.1.8[b][c] SR 3.3.1.11[b][c] SR 3.3.1.16	25% RTP
3. Power Range Neutron Flux Rate					
a. High Positive Rate	1,2	4	E	SR 3.3.1.7[b][c] SR 3.3.1.11[b][c] SR 3.3.1.16	4% RTP with time constant ≥ 2 sec
b. High Negative Rate	1,2	4	E	SR 3.3.1.7[b][c] SR 3.3.1.11[b][c] SR 3.3.1.16	4% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1[d], 2[e]	2	F,G	SR 3.3.1.1 SR 3.3.1.8[b][c] SR 3.3.1.11[b][c]	25% RTP

(continued)

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and as-left tolerances is specified in WCAP-17746-P.
(d) Below the P-10 (Power Range Neutron Flux) interlock.
(e) Above the P-6 (Intermediate Range Neutron Flux) interlock.

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	NOMINAL TRIP SETPOINT
5. Source Range Neutron Flux	2[f]	2	I,J	SR 3.3.1.1 SR 3.3.1.8[b][c] SR 3.3.1.11[b][c]	10^5 cps
	3[a], 4[a], 5[a]	2	J,K	SR 3.3.1.1 SR 3.3.1.7[b][c] SR 3.3.1.11[b][c]	10^5cps
6. Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7[b][c] SR 3.3.1.10[b][c] SR 3.3.1.16	Refer to Note 1 (Page 3.3-19)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7[b][c] SR 3.3.1.10[b][c] SR 3.3.1.16	Refer to Note 2 (Page 3.3-20)
8. Pressurizer Pressure					
a. Low	1[g]	4	M	SR 3.3.1.1 SR 3.3.1.7[b][c] SR 3.3.1.10[b][c] SR 3.3.1.16	1940 psig
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7[b][c] SR 3.3.1.10[b][c] SR 3.3.1.16	2385 psig
9. Pressurizer Water Level - High	1[g]	3	M	SR 3.3.1.1 SR 3.3.1.7[b][c] SR 3.3.1.10[b][c]	92% of instrument span

(continued)

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and as-left tolerances is specified in WCAP-17746-P.
(f) Below the P-6 (Intermediate Range Neutron Flux) interlock.
(g) Above the P-7 (Low Power Reactor Trips Block) interlock.

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	NOMINAL TRIP SETPOINT
10. Reactor Coolant Flow - Low	1[g]	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7[b][c] SR 3.3.1.10[b][c] SR 3.3.1.16	90% of indicated loop flow
11. Not Used.					
12. Undervoltage RCPs	1[g]	2/bus	M	SR 3.3.1.9 SR 3.3.1.10[b][c] SR 3.3.1.16	10578 Vac
13. Underfrequency RCPs	1[g]	2/bus	M	SR 3.3.1.9 SR 3.3.1.10[b][c] SR 3.3.1.16	57.2 Hz
14. Steam Generator (SG) Water Level Low-Low [l]	1,2	4 per gen	E	SR 3.3.1.1 SR 3.3.1.7[b][c] SR 3.3.1.10[b][c] SR 3.3.1.16	23.5% of Narrow Range Instrument Span
15. Not Used.					
16. Turbine Trip					
a. Low Fluid Oil Pressure	1[j]	3	O	SR 3.3.1.10[b][c] SR 3.3.1.15	590 psig
b. Turbine Stop Valve Closure	1[j]	4	P	SR 3.3.1.10 SR 3.3.1.15	≥ 1% open
17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	Q	SR 3.3.1.14	NA

(continued)

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and as-left tolerances is specified in WCAP-17746-P.
(g) Above the P-7 (Low Power Reactor Trips Block) interlock.
(l) The applicable MODES for these channels are more restrictive in Table 3.3.2-1. (See Function 6.d.)
(j) Above the P-9 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 4 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	NOMINAL TRIP SETPOINT
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2[f]	2	S	SR 3.3.1.11 SR 3.3.1.13	1.0E-10 amps
b. Low Power Reactor Trips Block, P-7	1	1 per train	T	SR 3.3.1.5	NA
c. Power Range Neutron Flux, P-8	1	4	T	SR 3.3.1.11 SR 3.3.1.13	48% RTP
d. Power Range Neutron Flux, P-9	1	4	T	SR 3.3.1.11 SR 3.3.1.13	50% RTP
e. Power Range Neutron Flux, P-10	1,2	4	S	SR 3.3.1.11 SR 3.3.1.13	10% RTP
f. Turbine Impulse Pressure, P-13	1	2	T	SR 3.3.1.10 SR 3.3.1.13	10% turbine power
19. Reactor Trip Breakers (RTB) [k]	1,2	2 trains	R	SR 3.3.1.4	NA
	3[a], 4[a], 5[a]	2 trains	C	SR 3.3.1.4	NA
20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms [k]	1,2	1 each per RTB	U	SR 3.3.1.4	NA
	3[a], 4[a], 5[a]	1 each per RTB	C	SR 3.3.1.4	NA
21. Automatic Trip Logic	1,2	2 trains	Q	SR 3.3.1.5	NA
	3[a], 4[a], 5[a]	2 trains	C	SR 3.3.1.5	NA

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(f) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(k) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \leq \Delta T_O \left\{ K_1 - K_2 \frac{(1+\tau_4 s)}{(1+\tau_5 s)}\left[T\left(\frac{1}{(1+\tau_6 s)}\right) - T' \right] + K_3(P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated loop ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T′ is the nominal T_{avg} (T_{ref} from Rod Control) at RTP, ≤ *.

P is the measured pressurizer pressure, psig.
P′ is the nominal RCS operating pressure ≥ * psig.

K_1 = *	K_2 = * /°F	K_3 = * /psig
τ_1 = * sec	τ_2 = * sec	τ_3 = * sec
τ_4 = * sec	τ_5 = * sec	τ_6 = * sec

$f_1(\Delta I)$ =	* { * % + (q_t - q_b)}	when q_t - q_b < * % RTP
	0% of RTP	when * % RTP ≤ q_t - q_b ≤ * % RTP
	* {(q_t - q_b) - * % }	when q_t - q_b > * % RTP

where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and q_t + q_b is the total THERMAL POWER in percent RTP.

The values denoted with * are specified in the COLR.

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

Note 2: Overpower ΔT

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \leq \Delta T_O \left\{ K_4 - K_5 \frac{(\tau_7 s)}{(1+\tau_7 s)}\left(\frac{1}{1+\tau_6 s}\right) T - K_6 \left[T \frac{1}{(1+\tau_6 s)} - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated loop ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T″ is the nominal T_{avg} (T_{ref} from Rod Control) at RTP, ≤ * °F.

K_4 = *	K_5 = * /°F for increasing T_{avg} * /°F for decreasing T_{avg}	K_6 = * /°F when T > T″ * /°F when T ≤ T″
τ_1 = * sec	τ_2 = * sec	τ_3 = * sec
τ_6 = * sec	τ_7 = * sec	
$f_2(\Delta I)$ = *		

The values denoted with * are specified in the COLR.

Table 3.3.2-1 (page 1 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	NOMINAL TRIP SETPOINT
1. Safety Injection					
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
c. Containment Pressure - High 1	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.5[b][c] SR 3.3.2.9[b][c] SR 3.3.2.10	3.5 psig
d. Pressurizer Pressure - Low	1,2,3[a]	4	D	SR 3.3.2.1 SR 3.3.2.5[b][c] SR 3.3.2.9[b][c] SR 3.3.2.10	1830 psig
e. Steam Line Pressure Low	1,2,3[a]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5[b][c] SR 3.3.2.9[b][c] SR 3.3.2.10	615 psig[d]
2. Containment Spray					
a. Manual Initiation	1,2,3,4	2 per train, 2 trains	B	SR 3.3.2.8	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
c. Containment Pressure High - 3	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5[b][c] SR 3.3.2.9[b][c] SR 3.3.2.10	27.0 psig

(continued)

(a) Above the P-11 (Pressurizer Pressure) interlock and below P-11 unless the Function is blocked.
(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and as-left tolerances is specified in WCAP-17746-P.
(d) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.

Table 3.3.2-1 (page 2 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	NOMINAL TRIP SETPOINT
3. Containment Isolation					
a. Phase A Isolation					
(1) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
b. Phase B Isolation					
(1) Manual Initiation	1,2,3,4	2 per train, 2 trains	B	SR 3.3.2.8	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
(3) Containment Pressure - High 3	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5[(b)(c)] SR 3.3.2.9[(b)(c)] SR 3.3.2.10	27.0 psig
4. Steam Line Isolation					
a. Manual Initiation	1,2[(i)], 3[(i)]	2	F	SR 3.3.2.8	NA
b. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2[(i)], 3[(i)]	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA

(continued)

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and as-left tolerances is specified in WCAP-17746-P.

(i) Except when all MSIVs are closed and de-activated; and all MSIV bypass valves are closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves.

Table 3.3.2-1 (page 3 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	NOMINAL TRIP SETPOINT
4. Steam Line Isolation (continued)					
c. Automatic Actuation Logic (MSFIS)	1,2[l], 3[l]	2 trains	G	SR 3.3.2.6	NA
d. Containment Pressure - High 2	1,2[i], 3[i]	3	D	SR 3.3.2.1 SR 3.3.2.5[b][c] SR 3.3.2.9[b][c] SR 3.3.2.10	17.0 psig
e. Steam Line Pressure					
(1) Low	1,2[i],3[a][i]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5[b][c] SR 3.3.2.9[b][c] SR 3.3.2.10	615 psig[d]
(2) Negative Rate - High	3[g][i]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5[b][c] SR 3.3.2.9[b][c] SR 3.3.2.10	100[h] psi
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2[j],3[j]	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
b. Automatic Actuation Logic (MSFIS)	1,2[k],3[k]	2 trains	G	SR 3.3.2.6	NA

(continued)

(a) Above the P-11 (Pressurizer Pressure) Interlock and below P-11 unless the Function is blocked.
(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and as-left tolerances is specified in WCAP-17746-P.
(d) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.
(g) Below the P-11 (Pressurizer Pressure) Interlock; however, may be blocked below P-11 when safety injection on low steam line pressure is not blocked.
(h) Time constant utilized in the rate/lag controller is ≥ 50 seconds.
(i) Except when all MSIVs are closed and de-activated; and all MSIV bypass valves are closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves.
(j) Except when all MFIVs are closed and de-activated; and all MFRVs are closed and de-activated or closed and isolated by a closed manual valve; and all MFRV bypass valves are closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves.
(k) Except when all MFIVs are closed and de-activated.
(l) Except when all MSIVs are closed and de-activated.

Table 3.3.2-1 (page 4 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	NOMINAL TRIP SETPOINT
5. Turbine Trip and Feedwater Isolation (continued)					
c. SG Water Level -High High (P-14)	1,2[j]	4 per SG	I	SR 3.3.2.1 SR 3.3.2.5[b][c] SR 3.3.2.9[b][c] SR 3.3.2.10	78% of Narrow Range Instrument Span
d. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
6. Auxiliary Feedwater					
a. Manual Initiation	1,2,3	1 per pump	O	SR 3.3.2.8	NA
b. Automatic Actuation Logic and Actuation Relays (Solid State Protection System)	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
c. Automatic Actuation Logic and Actuation Relays (Balance of Plant ESFAS)	1,2,3	2 trains	N	SR 3.3.2.3	NA
d. SG Water Level Low - Low	1,2,3	4 per SG	D	SR 3.3.2.1 SR 3.3.2.5[b][c] SR 3.3.2.9[b][c] SR 3.3.2.10	23.5% of Narrow Range Instrument Span
e. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
f. Loss of Offsite Power	1,2,3	2 trains	P	SR 3.3.2.7 SR 3.3.2.10	NA
g. Trip of all Main Feedwater Pumps	1	2 per pump	J	SR 3.3.2.8	NA
h. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	1,2,3	3	M	SR 3.3.2.1 SR 3.3.2.9[b][c] SR 3.3.2.10 SR 3.3.2.12[b][c]	21.6 psia

(continued)

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and as-left tolerances is specified in WCAP-17746-P.

(j) Except when all MFIVs are closed and de-activated; and all MFRVs are closed and de-activated or closed and isolated by a closed manual valve; and all MFRV bypass valves are closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves.

Table 3.3.2-1 (page 5 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	NOMINAL TRIP SETPOINT
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
b. Refueling Water Storage Tank (RWST) Level - Low Low	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.5[b][c] SR 3.3.2.9[b][c] SR 3.3.2.10	36.0% of instrument span
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
8. ESFAS Interlocks					
a. Reactor Trip, P-4[m]	1,2,3	2 per train, 2 trains	F	SR 3.3.2.11	NA
b. Pressurizer Pressure, P-11	1,2,3	3	L	SR 3.3.2.5 SR 3.3.2.9	1970 psig

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodology used to determine the as-found and as-left tolerances is specified in WCAP-17746-P.

(m) The functions of the Reactor Trip, P-4 interlock required to meet the LCO are:
- Trips the main turbine – MODES 1 and 2
- Isolates MFW with coincident low T_{avg} – MODES 1 and 2
- Allows manual block of the automatic reactuation of SI after a manual reset of SI – MODES 1, 2, and 3
- Prevents opening of MFIVs if closed on SI or SG Water Level – High High – MODES 1, 2, and 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Not Used.	
SR 3.3.5.2	----------NOTE---------- Verification of time delays is not required. ---------- Perform TADOT.	31 days
SR 3.3.5.3	Perform CHANNEL CALIBRATION with nominal Trip Setpoint and Allowable Value as follows: a. Loss of voltage Nominal Trip Setpoint 83V, 120V bus with a time delay of 1.0 sec. b. Degraded voltage Nominal Trip Setpoint 108.7V, 120V bus with a time delay of 119 sec.	18 months
SR 3.3.5.4	Verify LOP DG Start ESF RESPONSE TIMES are within limits.	18 months on a STAGGERED TEST BASIS

3.4 REACTOR COOLANT SYSTEM (RCS)

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

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

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

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

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

APPLICABILITY: MODE 1.

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

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

b. THERMAL POWER step > 10% RTP.

ACTIONS

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

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

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

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

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

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

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

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

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code."

5. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores."

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

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

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

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

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

(continued)

Proposed TS Bases Changes (for information only)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor ~~pressure~~ ~~coolant~~ boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

pressure

coolant

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and

BASES

APPLICABLE SAFETY ANALYSES (continued)

hence valve size requirements and lift settings, is a turbine trip without a direct reactor trip.

Cases with and without pressurizer spray and PORVs are analyzed. Safety valves on the secondary side are assumed to open when the steam pressure reaches the safety valve settings. Main feedwater supply is lost at the time of turbine trip ~~and the Auxiliary Feedwater System supplies feedwater flow to ensure adequate residual and heat removal capability~~.

The Reactor Trip System Allowable Values in Table 3.3.1-1, together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization. The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

a. Pressurizer power operated relief valves (PORVs);

b. Steam Generator Atmospheric Relief Valves (ARVs);

c. Condenser Steam Dump valves;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray valves.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The SL on maximum allowable RCS pressure is 2735 psig.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because of the plant conditions making it unlikely that the RCS can be pressurized.

BASES

APPLICABLE SAFETY ANALYSES (continued)

with respect to potential fuel damage ~~before a reactor trip occurs,~~ is a guillotine break of a main steam line inside containment initiated at the end of core life with RCS T_{avg} equal to 557°F. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The startup of an inactive RCP is administratively precluded in MODES 1 and 2. In MODE 3, the startup of an inactive RCP can not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

The ejection of a control rod rapidly adds reactivity to the reactor core reactor core causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

BASES

BACKGROUND (continued)

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The USAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3).

(part-power conditions) or zero (full-power conditions)

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal ~~transient from either zero (Ref. 2) or RTP~~, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

transients from either subcritical or at-power conditions

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the BOC or EOC life. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. An EOC measurement is conducted at conditions when the RCS boron concentration reaches a boron concentration equivalent to 300 ppm at an equilibrium, all rods out, RTP condition. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.2 (continued)

2. If the 300 ppm Surveillance limit is exceeded, it is possible that the EOC limit on MTC could be reached before the planned EOC. Because the MTC changes slowly with core depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the EOC limit.

3. The Surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is less negative than the 60 ppm Surveillance limit, the EOC limit will not be exceeded because of the gradual manner in which MTC changes with core burnup

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.

2. USAR, Chapter 15.

3. ~~NSAG-007, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station."~~

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

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28.
2. 10 CFR 50.46.
3. USAR, Chapter 15.
4. USAR, Section 4.3.1.5.
5. ~~NSAG-007, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station."~~

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.4 (continued)

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

3. Regulatory Guide 1.68, Revision 2, August, 1978.

4. ~~NSAG-007, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station."~~

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

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during Anticipated Operational Occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

INSERT 1

~~The LSSS, defined in this specification as the Allowable Values, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).~~

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the ~~(DNBR) limit;~~ SL value to prevent departure from nucleate boiling (DNB);

2. Fuel centerline melt shall not occur; and

3. The RCS pressure Safety Limit of 2735 psig shall not be exceeded.

Operation within the limits of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 20 and 10 CFR 100 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

INSERT 1

Technical Specifications are required by 10 CFR 50.36 to include LSSS for variables that have significant safety functions. LSSS are defined by the regulation as "Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a protective action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

The Nominal Trip Setpoint (NTSP) specified in Table 3.3.1-1 is a predetermined setting for a protection channel chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded. As such, the NTSP accounts for uncertainties in setting the channel (e.g., calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the NTSP ensures that SLs are not exceeded. Therefore, the NTSP meets the definition of an LSSS (Ref. 15).

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as "...being capable of performing its safety functions(s)." Relying solely on the NTSP to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protection channel setting during a surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the NTSP due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the NTSP and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protection channel. Therefore, the channel would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the channel within the established as-left tolerance around the NTSP to account for further drift during the next surveillance interval.

BASES

BACKGROUND (continued)

The RTS instrumentation is segmented into four distinct but interconnected modules as described in USAR, Chapter 7 (Ref. 1), and as identified below:

1. Field transmitters or process sensors: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured;

2. Signal Process Control and Protection System, including 7300 Process Protection System, Nuclear Instrumentation System (NIS), field contacts, and protection channel sets: provides signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system ~~devices~~ channels, and control board/control room/miscellaneous indications;

3. Solid State Protection System (SSPS), including input, logic, and output bays: initiates proper unit shutdown and/or ESF actuation in accordance with the defined logic, which is based on the bistable outputs from the signal process control and protection system; and

4. Reactor trip switchgear, including reactor trip breakers (RTBs) and bypass breakers: provides the means to interrupt power to the control rod drive mechanisms (CRDMs) and allows the rod cluster control assemblies (RCCAs), or "rods," to fall into the core and shut down the reactor. The bypass breakers allow testing of the RTBs at power.

Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. To account for the calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the ~~Trip Setpoint and Allowable Values~~ NTSP. The OPERABILITY of each transmitter or sensor ~~can be evaluated when its "as found" calibration data are compared against its documented acceptance criteria.~~ is determined by "as-found" calibration data evaluated during the CHANNEL CALIBRATION based on the criteria defined in WCAP-17746-P, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station" (Ref. 5) and WCAP-17602-P, "Westinghouse Setpoint Calculations for the Wolf Creek Generating Station Control, Protection, and Indication Systems," (Ref. 16). The OPERABILITY of each transmitter or sensor may also be determined by qualitative assessment of the field transmitter or sensor as related to the channel behavior observed during performance of the CHANNEL CHECK.

the

Analytical Limits are defined in USAR Chapter 15 (Ref. 2).

BASES

BACKGROUND (continued)

NTSPs derived from Analytical Limits

Signal Process Control and Protection System

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with ~~setpoints~~ established by safety analyses. If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the SSPS for decision evaluation. Channel separation is maintained up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS, while others provide input to the SSPS, the main control board, the unit computer, and one or more control systems.

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function is still OPERABLE with a two-out-of-two logic. If one channel fails, such that a partial Function trip occurs, a trip will not occur and the Function is still OPERABLE with a one-out-of-two logic.

Generally, if a parameter is used for input to the SSPS and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Again, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-279-1971 (Ref. 3). The actual number of channels required for each unit parameter is specified in Reference 1.

Two logic channels are required to ensure no single random failure of a logic channel will disable the RTS. The logic channels are designed such that testing required while the reactor is at power may be accomplished without causing a trip. Provisions to allow removing logic channels from service during maintenance are unnecessary because of the logic system's designed reliability.

BASES

BACKGROUND (continued)

The methodology described in Reference 5 incorporates all of the known uncertainties applicable to each channel. The magnitudes of these uncertainties are factored into the determination of each NTSP. Field sensors and signal processing equipment for the associated channels are assumed to operate within the allowances of these uncertainty magnitudes. The as-left and as-found tolerance band methodology is provided in Reference 5. Reference 16 provides the as-left and as-found tolerance values.

The NTSP is the value at which the bistable is set and is the expected value to be achieved during calibration. The NTSP value is the LSSS and ensures the safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on stated channel uncertainties. Any bistable is considered to be properly adjusted when the as-left NTSP value is within the as-left tolerance band for CHANNEL CALIBRATION uncertainty allowance (i.e., ± rack calibration accuracy). The NTSP value is therefore considered a "nominal" value (i.e., expressed as a value without inequalities) for the purposes of COT and CHANNEL CALIBRATION.

NTSPs, in conjunction with the use of as-found and as-left tolerances, ensure that SLs are not violated during AOOs (and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed).

Trip Setpoints ~~and Allowable Values~~

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the two-sided tolerance band for channel accuracy ~~(typically ±15mV)~~.

calculation

the NTSPs specified in Table 3.3.1-1

The Trip Setpoints ~~listed in Table B 3.3.1-1 and~~ used in the bistables are based on the analytical limits stated in Reference 2. The ~~selection~~ of ~~these Trip Setpoints~~ is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RTS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 4), the ~~Allowable Values~~ specified in Table 3.3.1-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the ~~Trip Setpoints~~, including their explicit uncertainties, is provided in ~~the "Wolf Creek Nuclear Safety Analysis Setpoint Methodology for the Reactor Protection System" (Ref. 5).~~ The actual nominal Trip Setpoint ~~entered~~ into the bistable is more ~~conservative~~ than that ~~specified by~~ the Allowable Value to account for changes in random ~~measurement~~ errors detectable by a COT. One example of such ~~a change~~ in measurement ~~error~~ is drift during the surveillance ~~interval~~. If the measured setpoint does not ~~exceed~~ the ~~Allowable Value, the bistable is considered OPERABLE.~~

NTSPs

WCAP-17746-P, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station," (Ref. 5) and WCAP-17602-P, "Westinghouse Setpoint Calculations for the Wolf Creek Generating Station Control, Protection, and Indication Systems," (Ref. 16).

~~Setpoints in accordance with the Allowable Value ensure that design limits are not violated during AOOs (and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed). Note that in the accompanying LCO 3.3.1, the Allowable Values of Table 3.3.1-1 are the LSSS.~~

Each channel of the process control equipment can be tested on line to verify that the signal or setpoint accuracy is within the specified allowance requirements. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SRs section.

~~The Allowable Values listed in Table 3.3.1-1 are based on the methodology, which incorporates all of the known uncertainties applicable for each channel. The essential elements of the methodology are~~

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BACKGROUND

~~Trip Setpoints and Allowable Values (continued)~~

~~described in Reference 10 and subsequently updated in Reference 5. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.~~

Solid State Protection System

The SSPS equipment is used for the decision logic processing of outputs from the signal processing equipment bistables. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide reactor trip and/or ESF actuation for the unit. If both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements. The system has been designed to trip in the event of a loss of power, directing the unit to a safe shutdown condition.

The SSPS performs the decision logic for actuating a reactor trip or ESF actuation, generates the electrical output signal that will initiate the required trip or actuation, and provides the status, permissive, and annunciator output signals to the main control room of the unit.

The bistable outputs from the signal processing equipment are sensed by the SSPS equipment and combined into logic matrices that represent combinations indicative of various unit upset and accident transients. If a required logic matrix combination is completed, the system will initiate a reactor trip or send actuation signals via master and slave relays to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

Reactor Trip Switchgear

The RTBs are in the electrical power supply line from the control rod drive motor generator set power supply to the CRDMs. Opening of the RTBs interrupts power to the CRDMs, which allows the shutdown rods and control rods to fall into the core by gravity. Each RTB is equipped with a bypass breaker to allow testing of the RTB while the unit is at power.

BASES

BACKGROUND

Reactor Trip Switchgear (continued)

During normal operation the output from the SSPS is a voltage signal that energizes the undervoltage coils in the RTBs and bypass breakers, if in use. When the required logic matrix combination is completed, the SSPS output voltage signal is removed, the undervoltage coils are de-energized, the breaker trip lever is actuated by the de-energized undervoltage coil, and the RTBs and bypass breakers are tripped open. This allows the shutdown rods and control rods to fall into the core. In addition to the de-energization of the undervoltage coils, each reactor trip breaker is also equipped with a shunt trip device that is energized to trip the breaker open upon receipt of a reactor trip signal from the SSPS. Either the undervoltage coil or the shunt trip mechanism is sufficient by itself, thus providing a diverse trip mechanism.

The decision logic matrix Functions are described in the functional diagrams included in Reference 1. In addition to the reactor trip or ESF actuation, these diagrams also describe the various "permissive interlocks" that are associated with unit conditions. Each SSPS train has a built in testing device that can test the decision logic matrix Functions and the actuation devices while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The RTS functions to ~~maintain~~ preserve the ~~applicable Safety Limits~~ SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

Each of the analyzed accidents and transients can be detected by one or more RTS Functions. The accident analysis described in Reference 2 takes credit for most RTS trip Functions. RTS trip Functions that are retained yet not specifically credited in the accident analysis are ~~qualitatively~~ implicitly credited in the safety analysis and the NRC staff approved licensing basis for the unit. These RTS trip Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. They may also serve as backups to RTS trip Functions that were credited in the accident analysis.

INSERT 2

~~The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.~~

INSERT 2

Permissive and interlock setpoints allow the blocking of trips during plant startups, and restoration of trips when the permissive conditions are not satisfied, but they are not explicitly modeled in the safety analyses. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventive or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative starting assumptions for the accident analysis, they are generally considered as nominal values without regard to measurement accuracy.

The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 to be OPERABLE. A channel is OPERABLE if the as-found setpoint is within the as-found tolerance of the NTSP specified in Table 3.3.1-1 during a CHANNEL CALIBRATION or CHANNEL OPERATIONAL TEST (COT). In this manner, the actual setting of the channel (NTSP) will ensure that a SL is not exceeded at any given point of time as long as the channel has not drifted beyond expected tolerances during the surveillance interval. Note that, although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left tolerance, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned.

If the actual setting of the channel is found to be conservative with respect to the NTSP but is beyond the as-found tolerance band, the channel is OPERABLE but degraded. The degraded condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTSP (within the allowed tolerance), and evaluating the channel response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation.

A trip setpoint may be set more conservative than the NTSP as necessary in response to plant conditions. However, in this case, the OPERABILITY of this instrument must be verified based on the field setting and not the NTSP. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY (continued)

The LCO generally requires OPERABILITY of four or three channels in each instrumentation Function, two channels of Manual Reactor Trip in each logic Function, and two trains in each Automatic Trip Logic Function. Four OPERABLE instrumentation channels in a two-out-of-four configuration are ~~generally~~ required when one RTS channel is also used as a control system input. This configuration accounts for the possibility of the shared channel failing in such a manner that it creates a transient that requires RTS action. In this case, the RTS will still provide protection, even with a random failure of one of the other three protection channels. Three OPERABLE instrumentation channels in a two-of-three configuration are generally required when there is no potential for control system and protection system interaction that could simultaneously create a need for RTS trip and disable one RTS channel. The two-out-of-three and two-out-of-four logic configurations allow one channel to be tripped for maintenance or surveillance testing without causing a reactor trip. In cases where an inoperable channel is placed in the tripped condition indefinitely to satisfy the Required Action of an LCO the logic configurations are reduced to one-out-of-two and one-out-of-three where tripping of an additional channel, for any reason, would result in a reactor trip. To allow for surveillance testing or setpoint adjustment of other channels while in this condition, several Required Actions allow the inoperable channel to be bypassed. Bypassing the inoperable channel creates a two-out-of-two or two-out-of-three logic configuration allowing a channel to be tripped for testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

Reactor Trip System Functions

The safety analyses and OPERABILITY requirements applicable to each RTS Function are discussed below:

1. Manual Reactor Trip

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time by using either of two reactor trip switches in the control room. A Manual Reactor Trip accomplishes the same results as any one of the automatic trip Functions. It is used by the reactor operator to shut down the reactor whenever any parameter is rapidly trending toward its Trip Setpoint.

The LCO requires two Manual Reactor Trip channels to be

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Power Range Neutron Flux - High (continued)

levels, against a positive reactivity excursion leading to DNB during power operations. These can be caused by rod withdrawal or reductions in RCS temperature.

The LCO requires all four of the Power Range Neutron Flux - High channels to be OPERABLE. ~~The Trip Setpoint is ≤ 109% RTP.~~

In MODE 1 or 2, when a positive reactivity excursion could occur, the Power Range Neutron Flux - High trip must be OPERABLE. This Function will terminate the reactivity excursion and shut down the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the NIS power range detectors do not provide neutron level indication in this range. In these MODES, the Power Range Neutron Flux - High do not have to be OPERABLE because the reactor is shut down and reactivity excursions into the power range are extremely unlikely. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6.

b. Power Range Neutron Flux - Low

The LCO requirement for the Power Range Neutron Flux - Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions.

The LCO requires all four of the Power Range Neutron Flux - Low channels to be OPERABLE. ~~The Trip Setpoint is ≤ 25% RTP.~~

In MODE 1, below the Power Range Neutron Flux (P-10 setpoint), and in MODE 2, the Power Range Neutron Flux - Low trip must be OPERABLE. This Function may be manually blocked by the operator when two out of four power range channels are greater than 10% RTP (P-10 setpoint). This Function is automatically unblocked when three out of four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux - High trip Function.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Power Range Neutron Flux - High Negative Rate (continued)

result in an unconservative local DNBR. DNBR is defined as the ratio of the heat flux required to cause a DNB at a particular location in the core to the local heat flux. The DNBR is indicative of the margin to DNB. No credit is taken for the operation of this Function for those rod drop accidents in which the local DNBRs will be greater than the limit.

The LCO requires all four Power Range Neutron Flux-High Negative Rate channels to be OPERABLE. ~~The Trip Setpoint is ≤ 4% RTP with a time constant ≥ 2 seconds.~~

In MODE 1 or 2, when there is potential for a multiple rod drop accident to occur, the Power Range Neutron Flux - High Negative Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux - High Negative Rate trip Function does not have to be OPERABLE because the core is not critical and DNB is not a concern.

4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition. This trip Function provides redundant protection to the Power Range Neutron Flux - Low Setpoint trip Function. The NIS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors do not provide any input to control systems. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. ~~The Trip Setpoint is ≤ 25% RTP.~~

Because this trip Function is important only during startup, there is generally no need to disable channels for testing while the

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. Intermediate Range Neutron Flux (continued)

Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

In MODE 1 below the P-10 setpoint, and in MODE 2, above the P-6 setpoint when there is a potential for an uncontrolled RCCA bank rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux - High Setpoint trip and the Power Range Neutron Flux - High Positive Rate trip provide core protection for a rod withdrawal accident. In Modes 2 (below the P-6 setpoint), the Source Range Neutron Flux trip Function provides core protection for reactivity accidents. In MODE 3, 4, 5, or 6, the Intermediate Range Neutron Flux trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against unacceptable positive reactivity additions.

5. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition. This trip Function provides redundant protection to the Power Range Neutron Flux - Low trip Function. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled withdrawal of rods. The NIS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RTS automatic protection function required in MODES 3, 4, and 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. Therefore, the functional capability at the specified Trip Setpoint is assumed to be available.

The LCO requires two channels of Source Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. ~~The Trip Setpoint is ≤ 1.0 E5 cps.~~ The outputs of the Function to RTS logic are not required OPERABLE in MODE 6 or when all rods are fully inserted and the Rod Control System is incapable of rod withdrawal.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

8. Pressurizer Pressure

Four Pressurizer Pressure channels provide input to the Pressurizer Pressure - High and - Low trips and the Overtemperature ΔT trip. The Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System; thus, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

a. Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

The LCO requires four channels of Pressurizer Pressure-Low to be OPERABLE. ~~The Trip Setpoint is ≥ 1940 psig.~~

In MODE 1, when DNB is a major concern, the Pressurizer Pressure - Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock (NIS power range P-10 or turbine impulse pressure greater than approximately 10% of full power equivalent (P-13). On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, there is insufficient heat production to generate DNB conditions.

b. Pressurizer Pressure - High

The Pressurizer Pressure - High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Pressurizer Pressure – High (continued)

The LCO requires four channels of the Pressurizer Pressure - High to be OPERABLE. ~~The Trip Setpoint is ≤ 2385 psig.~~

NTSP

The Pressurizer Pressure - High ~~Allowable Value~~ is selected to be below the pressurizer safety valve actuation pressure and above the power operated relief valve (PORV) setting. This setting minimizes challenges to safety valves while avoiding unnecessary reactor trip for those pressure increases that can be controlled by the PORVs.

In MODE 1 or 2, the Pressurizer Pressure - High trip must be OPERABLE to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure - High trip Function does not have to be OPERABLE because transients that could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate unit conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when the temperature of one or more RCS loops is below 368°F.

9. Pressurizer Water Level - High

The Pressurizer Water Level - High trip Function provides a backup signal for the Pressurizer Pressure - High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The LCO requires three channels of Pressurizer Water Level - High to be OPERABLE. ~~The Trip Setpoint is ≤ 92% of instrument span.~~ The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before reactor high pressure trip.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Pressurizer Water Level - High (continued)

In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level - High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

10. Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-7 setpoint, the reactor trip on low flow in two or more RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow - Low channels per loop to be OPERABLE in MODE 1 above P-7. ~~The Trip Setpoint is ≥ 89.9% of loop design flow (loop design flow - 90,324 gpm).~~

In MODE 1, above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in the core because of the higher power level. In MODE 1, below the P-8 setpoint and above the P-7 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip because of the lower power level and the greater margin to the design limit DNBR. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since there is insufficient heat production to generate DNB conditions.

11. Not Used.

12. Undervoltage Reactor Coolant Pumps

The Undervoltage RCP reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. There is one potential transformer (PT), with a primary to secondary ratio of 14400:120,

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

12. Undervoltage Reactor Coolant Pumps (continued)

connected in parallel with the 13.8 kV power supply to each RCP motor at the motor side of the supply breaker. Each PT secondary side is connected to an undervoltage relay and time delay relay, as well as a separate underfrequency relay. The undervoltage relays provide output signals to the SSPS which trips the reactor, if permissive P-7 is satisfied (i.e. greater than 10% of rated thermal power), when the voltage at one out of two RCP motors on both buses drops below ~~10578 Vac~~ the NTSP. The time delay relay prevents spurious trips caused by transient voltage perturbations. This trip Function will generate a reactor trip before the Reactor Coolant Flow - Low Trip Setpoint is reached.

The LCO requires two Undervoltage RCP channels per bus to be OPERABLE, for a total of four channels. ~~The Trip Setpoint is ≥ 10,578 Vac.~~

In MODE 1 above the P-7 setpoint, the Undervoltage RCP trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of RCP due to undervoltage are automatically blocked since the core is not producing sufficient power to generate DNB conditions. Above the P-7 setpoint, the reactor trip on Undervoltage RCPs is automatically enabled.

13. Underfrequency Reactor Coolant Pumps

The Underfrequency RCP reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. An adequate coastdown time is required so that reactor heat can be removed immediately after reactor trip. There is one potential transformer (PT), with a primary to secondary ratio of 14400:120, connected in parallel with the 13.8 kV power supply to each RCP motor at the motor side of the supply breaker. Each PT secondary side is connected to an undervoltage relay and time delay relay, as well as a separate underfrequency relay. The underfrequency relays provide output signals to the SSPS which trips the reactor, if permissive P-7 is satisfied (i.e. of greater than 10% of rated thermal power), when the frequency of one out of two RCP motors on both buses drops below ~~57.2 Hz~~ the NTSP. The time delay set on the underfrequency relay prevents spurious trips

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

13. Underfrequency Reactor Coolant Pumps (continued)

caused by transient frequency perturbations. This trip Function will generate a reactor trip before the Reactor Coolant Flow - Low Trip Setpoint is reached.

The LCO requires two Underfrequency RCP channels per bus to be OPERABLE, for a total of four channels. ~~The Trip Setpoint is ≥ 57.2 Hz.~~

In MODE 1 above the P-7 setpoint, the Underfrequency RCP trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of RCP due to underfrequency are automatically blocked since the core is not producing sufficient power to generate DNB conditions. Above the P-7 setpoint, the reactor trip on Underfrequency RCPs is automatically enabled.

14. Steam Generator Water Level - Low Low

The SG Water Level - Low Low trip Function ensures that protection is provided against a loss of heat sink and actuates the AFW System prior to uncovering the SG tubes. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low low level in any SG is indicative of a loss of heat sink for the reactor. The level transmitters provide input to the SG Level Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. This Function also performs the ESFAS function of starting the AFW pumps on low low SG level.

The LCO requires four channels of SG Water Level - Low Low per SG to be OPERABLE because these channels are shared between protection and control. ~~The Trip Setpoint for the SG Water Level Low - Low is ≥ 23.5% of narrow range instrument span.~~

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level - Low Low trip must be OPERABLE. The normal source of water for the SGs is provided by the Main Feedwater (MFW) pumps (not safety related). The MFW pumps are only in operation in MODE 1 or 2 above the point of adding heat. The

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

14. Steam Generator Water Level - Low Low (continued)

AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor. During normal startups and shutdowns (with RCS temperatures above 450° F), the motor driven Startup Feedwater pump provides feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level - Low Low Reactor trip Function does not have to be OPERABLE because the reactor is not operating or even critical (see Specification 3.3.2 for Applicability of SG Water Level - Low Low ESFAS Functions). Decay heat removal is accomplished by the Startup Feed pump or, if an AFAS is initiated, by the AFW System in MODE 3 and by the Residual Heat Removal (RHR) System in MODE 4, 5, or 6.

15. Not Used.

16. Turbine Trip

a. Turbine Trip - Low Fluid Oil Pressure

The Turbine Trip - Low Fluid Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. This trip Function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint, approximately 50% power, will not actuate a reactor trip. Three pressure ~~switches~~ monitor the control oil pressure in the Turbine Electrohydraulic Control System. A low pressure condition sensed by two-out-of-three pressure ~~switches~~ will actuate a reactor trip. These pressure ~~switches~~ do not provide any input to the control system. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure - High trip Function and RCS integrity is ensured by the pressurizer safety valves.

channels

channels

The LCO requires three channels of Turbine Trip - Low Fluid Oil Pressure to be OPERABLE in MODE 1 above P-9. ~~The Trip Setpoint is ≥ 590.0 psig.~~

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Turbine Trip - Low Fluid Oil Pressure (continued)

Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip, and the Turbine Trip - Low Fluid Oil Pressure trip Function does not need to be OPERABLE.

b. Turbine Trip - Turbine Stop Valve Closure

The Turbine Trip - Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. This trip Function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from power level below the P-9 setpoint, approximately 50% power, will not activate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will not and is not required to operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure - High trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Trip - Low Fluid Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RTS. If all four limit switches indicate that the stop valves are all closed, a reactor trip is initiated. The ~~LSSS Allowable Value~~ NTSP for this Function is set to assure channel trip occurs when the associated stop valve is completely closed.

The LCO requires four Turbine Trip - Turbine Stop Valve Closure channels, one per valve, to be OPERABLE in MODE 1 above P-9. All four channels must trip to cause reactor trip.

Below the P-9 setpoint, a load rejection can be accommodated by the Steam Dump System. In MODE 2, 3, 4, 5, or 6, there is no potential for a load rejection, and

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Turbine Trip - Turbine Stop Valve Closure (continued)

the Turbine Trip-Stop Valve Closure trip Function does not need to be OPERABLE.

17. Safety Injection Input from Engineered Safety Feature Actuation System

The SI Input from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any automatic signal that initiates SI. This is a condition of acceptability for the LOCA. However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

An NTSP is

~~Trip Setpoint and Allowable Values are~~ not applicable to this Function. The SI reactor trip input to SSPS logic is provided by ESFAS relay actuation. Therefore, there is no measurement signal with which to associate an LSSS.

The LCO requires two trains of SI Input from ESFAS to be OPERABLE in MODE 1 or 2.

A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical, and must be shut down in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical, and this trip Function does not need to be OPERABLE.

18. Reactor Trip System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current unit status. They back up operator actions to ensure protection system Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the interlock Functions do not need to be OPERABLE when the associated reactor trip functions are outside the applicable MODES. These are:

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when any NIS intermediate range channel goes approximately one decade above the minimum channel reading. If both channels drop below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 interlock ensures that the following Functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range. When the source range trip is blocked, the high voltage to the detectors is also removed; and

- on decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the NIS Source Range Neutron Flux reactor trip.

The LCO requires two channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint. ~~The Trip Setpoint is ≥ 1.0 E-10 amps.~~

Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip will be blocked, and this Function will no longer be necessary. In MODE 3, 4, 5, or 6, the P-6 interlock does not have to be OPERABLE because the NIS Source Range is providing core protection.

b. Low Power Reactor Trips Block, P-7

The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either the Power Range Neutron Flux, P-10, or the Turbine Impulse Chamber Pressure, P-13 interlock. The LCO requirement for the P-7 interlock ensures that the following Functions are performed:

(1) on increasing power, the P-7 interlock automatically enables reactor trips on the following Functions:

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Low Power Reactor Trips Block, P-7 (continued)

- Pressurizer Pressure - Low;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low (low flow in two or more RCS loops);
- Undervoltage RCPs; and
- Underfrequency RCPs.

These reactor trips are only required when operating above the P-7 setpoint (approximately 10% power). The reactor trips provide protection against violating the DNBR limit. Below the P-7 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

(2) on decreasing power, the P-7 interlock automatically blocks reactor trips on the following Functions:

- Pressurizer Pressure - Low;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low (low flow in two or more RCS loops);
- Undervoltage RCPs; and
- Underfrequency RCPs.

NTSPs

~~Allowable Values~~ are not applicable to the P-7 interlock because it is a logic Function and thus has no parameter with which to associate an LSSS.

The P-7 interlock is a logic Function with train and not channel identity. Therefore, the LCO requires one channel per train of Low Power Reactor Trips Block, P-7 interlock to be OPERABLE in MODE 1.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Low Power Reactor Trips Block, P-7 (continued)

The low power trips are blocked below the P-7 setpoint and unblocked above the P-7 setpoint. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the interlock performs its Function when power level drops below 10% power, which is in MODE 1.

c. Power Range Neutron Flux, P-8

The Power Range Neutron Flux, P-8 interlock is actuated at 48% power as determined by two-out-of-four NIS power range channels . The P-8 interlock automatically enables the Reactor Coolant Flow - Low reactor trip on low flow in one or more RCS loops on increasing power. The LCO requirement for this trip Function ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when greater than 48% power. On decreasing power, the reactor trip on low flow in any loop is automatically blocked.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1. ~~The Trip Setpoint is ≤ 48% RTP.~~

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

d. Power Range Neutron Flux, P-9

The Power Range Neutron Flux, P-9 interlock is actuated at 50% power as determined by two-out-of-four NIS power range channels. The LCO requirement for this Function ensures that the Turbine Trip - Low Fluid Oil Pressure and Turbine Trip -Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip will cause a load rejection beyond the capacities of the Steam Dump and Reactor Control Systems. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint, to minimize the transient on the reactor.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

d. Power Range Neutron Flux, P-9 (continued)

The LCO requires four channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1. ~~The Trip Setpoint is ≤ 50% RTP.~~

In MODE 1, a turbine trip could cause a load rejection beyond the capacities of the Steam Dump and Reactor Control Systems, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a load rejection beyond the capacities of the Steam Dump and Reactor Control Systems.

e. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at 10% power, as determined by two-out-of-four NIS power range channels. If power level falls below 10% RTP on 3 of 4 channels, Power Range Neutron Flux - Low reactor trip and the Intermediate Range Neutron Flux reactor trip and rod stop will be automatically unblocked. The LCO requirement for the P-10 interlock ensures that the following Functions are performed:

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip. Note that blocking the reactor trip also blocks the signal to prevent automatic and manual rod withdrawal;
- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux - Low reactor trip;
- on increasing power, the P-10 interlock automatically provides a backup signal to block the Source Range Neutron Flux reactor trip, and also to de-energize the NIS source range detectors;
- the P-10 interlock provides one of the two inputs to the P-7 interlock; and

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

e. Power Range Neutron Flux, P-10 (continued)

- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux - Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2. ~~The Trip Setpoint is 10% RTP.~~

OPERABILITY in MODE 1 ensures the Function is available to perform its decreasing power Functions in the event of a reactor shutdown. This Function must be OPERABLE in MODE 2 to ensure that core protection is provided during a startup or shutdown by the Power Range Neutron Flux - Low and Intermediate Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

greater than rated This

f. Turbine Impulse Pressure, P-13

The Turbine Impulse Pressure, P-13 interlock is actuated when the pressure in the first stage of the high pressure turbine is approximately 10% of the full power pressure. ~~The full power pressure corresponds to the first stage pressure at 100% RTP. The interlock~~ is determined by one-out-of-two pressure channels. The LCO requirement for this Function ensures that one of the inputs to the P-7 interlock is available.

The LCO requires two channels of Turbine Impulse Pressure, P-13 interlock to be OPERABLE in MODE 1. ~~The Trip Setpoint is ≤ 10% Turbine Power.~~

The Turbine Impulse Chamber Pressure, P-13 interlock must be OPERABLE when the turbine generator is operating. The interlock Function is not required OPERABLE in MODE 2, 3, 4, 5, or 6 because the turbine generator is not operating.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

21. Automatic Trip Logic (continued)

breaker while the unit is at power. The reactor trip signals generated by the RTS Automatic Trip Logic cause the RTBs and associated bypass breakers to open and shut down the reactor.

The LCO requires two trains of RTS Automatic Trip Logic to be OPERABLE. Having two OPERABLE channels ensures that random failure of a single logic channel will not prevent reactor trip.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

The RTS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

NTSP cannot be re-set to within the two-sided as-left tolerance band or the channel is not functioning as required

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.1-1.

In the event a channel's ~~Trip Setpoint is found nonconservative with respect to the Allowable Value~~, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Function channels provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function channels affected. When the Required Channels in Table ~~3.3-1~~ 3.3.1-1 are specified on a per loop, per SG, per bus, etc., basis, then the Condition may be entered separately for each loop, SG, bus, etc., as appropriate.

When the number of inoperable channels in a trip Function exceed those specified in one or other related Conditions associated with a trip Function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies to all RTS protection Functions. Condition A addresses the situation where one or more required channels or trains for one or more Functions are inoperable at the same time. The Required

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1 (continued)

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the power range channel output every 24 hours. If the calorimetric heat balance calculation results exceed the power channel output by more than + 2% RTP, the power range channel is not declared inoperable, but must be adjusted consistent with the calorimetric heat balance calculation results. If the power range channel output cannot be properly adjusted, the channel is declared inoperable.

If the calorimetric is performed at part-power (< 45% RTP), adjusting the power range channel indication in the increasing power direction will assure a reactor trip below the power range high safety analysis limit (SAL) in USAR Table 15.0-4 (≤ ~~118~~ 116.5% RTP) (Ref. 11). Making no adjustment to the power range channel in the decreasing power direction due to a part-power calorimetric assures a reactor trip consistent with the safety analyses.

This allowance does not preclude making indicated power adjustments, if desired, when the calorimetric heat balance calculation power is less than the power range channel output. To provide close agreement between indicated power and to preserve operating margin, the power range channels are normally adjusted when operating at or near full power during steady-state conditions. However, discretion must be exercised if the power range channel output is adjusted in the decreasing power direction due to a part-power calorimetric (< 45% RTP). This action may introduce a non-conservative bias at higher power levels which could delay an NIS reactor trip until power is above the power range SAL. The cause of the non-conservative bias is the decreased accuracy of the calorimetric at reduced power conditions.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

The primary error contributor to the instrument uncertainty for a secondary side power calorimetric measurement is the feedwater flow measurement, which is determined by a ΔP measurement across a feedwater venturi. While the measurement uncertainty remains constant in ΔP span as power decreases, when translated into flow the uncertainty increases as a square term. Therefore, a 1% flow error at 100% power can approach a 10% flow error at 30% RTP even though the ΔP error has not changed.

Thus, it is required to adjust the setpoint of the Power Range Neutron Flux – High bistables to ≤ 80% RTP: 1) prior to adjustment of the power range channel output in the decreasing power direction due to a part-power calorimetric below 45% RTP; or 2) for a post refueling startup. The evaluation of extended operation at part-power conditions concludes that the potential need to adjust the indication of the Power Range Neutron Flux in the decreasing power direction is quite small, primarily to address operation in the intermediate range about P-10 (nominally 10% RTP) to allow enabling of the Power Range Neutron Flux – Low setpoint and the Intermediate Range Neutron Flux reactor trips. Before the Power Range Neutron Flux – High bistables are reset to ≤ ~~109% RTP~~, the power range channel adjustment must be confirmed based on the NTSP a calorimetric performed at ≥ 45% RTP.

The Note to SR 3.3.1.2 clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP. A power level of 15% RTP is chosen based on plant stability, i.e., automatic rod control capability and the turbine generator synchronized to the grid. The 24 hour allowance after increasing THERMAL POWER above 15% RTP provides a reasonable time to attain a scheduled power plateau, establish the requisite conditions, perform the calorimetric measurement, and make any required adjustments in a controlled, orderly manner and without introducing the potential for extended operation at high power levels with instrumentation that has not been verified to be OPERABLE for subsequent use.

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate that a difference between the calorimetric heat balance calculation and the power range channel output of more than + 2% RTP is not expected in any 24 hour period.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the core power distribution, measured using either the movable incore detector system or the Power Distribution Monitoring System. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the core power distribution measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is not required to be performed until 72 hours after achieving equilibrium conditions with THERMAL POWER ≥ 75% RTP. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to obtain a core power distribution measurement. The SR is deferred until a scheduled testing plateau above 75% RTP is attained during a power ascension. During a typical power ascension, it is usually necessary to control the axial flux difference at lower power levels through control rod insertion. After equilibrium conditions are achieved at the specified power plateau, a core power distribution measurement must be taken and the required data collected. The data is typically analyzed and the appropriate excore calibrations completed within 48 hours after achieving equilibrium conditions. An additional time allowance of 24 hours is provided during which the effects of equipment failures may be remedied and any required re-testing may be performed.

The allowance of 72 hours after equilibrium conditions are attained at the testing plateau provides sufficient time to allow power ascensions and associated testing to be conducted in a controlled and orderly manner at conditions that provide acceptable results and without introducing the potential for extended operation at high power levels with instrumentation that has not been verified to be OPERABLE for subsequent use.

The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

The difference between the current "as-found" values and the previous test "as-left" values must be consistent with the drift allowance used in the setpoint methodology (Reference 5). The setpoint shall be left set consistent with the assumptions of the setpoint methodology described in Reference 5. The "as-found" and "as-left" values must also be recorded and trended for consistency with the assumptions of References 5 and 16.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 184 days.

A COT is performed on each required channel to ensure the channel will perform the intended Function.

~~Setpoints must be within the Allowable Values specified in Table 3.3.1-1.~~

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.7 (continued)

SR 3.3.1.7 is modified by a Note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for > 4 hours this Surveillance must be performed prior to 4 hours after entry into MODE 3. Note 2 requires that the quarterly COT for the source range instrumentation shall include verification by observation of the associated permissive annunciator window that the P-6 and P-10 interlocks are in their required state for the existing conditions.

The Frequency of 184 days is justified in Reference 13.

INSERT 3

SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, and it is modified by a Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit conditions. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed, e.g., by observation of the associated permissive annunciator window, within 184 days of the Frequencies prior to reactor startup, 12 hours after reducing power below P-10, and four hours after reducing power below P-6. The Frequency of "prior to startup" ensures this surveillance is performed prior to critical operations and applies to the source, intermediate and power range low instrument channels. The Frequency of "12 hours after reducing power below P-10" (applicable to intermediate and power range low channels) and "4 hours after reducing power below P-6" (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 184 days thereafter applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup, 12 hours after reducing power below P-10, and four hours after reducing power below P-6. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 for more than 12 hours or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 12 hour or the 4 hour limit. These time limits are reasonable, based

INSERT 3

The Surveillance Requirement is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also states that the methodology for calculating the as-left and the as-found tolerances is specified in WCAP-17746-P (Ref. 5).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.8 (continued)

on operating experience to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for the periods discussed above. The Frequency of 184 days is justified in Reference 13.

INSERT 3

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT and is performed every 92 days, as justified in Reference 6.

This SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to RCP undervoltage and underfrequency relays, setpoint verification requires elaborate bench calibration and is accomplished during the CHANNEL CALIBRATION.

SR 3.3.1.10

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The difference between the current "as-found" values and the previous test "as-left" values when trended must be consistent with the drift allowance used in the setpoint methodology (References 5 and 16).

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint methodology.

The Frequency of 18 months is based on the assumed calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable. This does not include verification of time delay relays. These are verified by response time testing per SR 3.3.1.16. Whenever an RTD is replaced in Functions 6 or 7, the next required CHANNEL CALIBRATION of the RTDs is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

INSERT 3

INSERT 3

The Surveillance Requirement is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also states that the methodology for calculating the as-left and the as-found tolerances is specified in WCAP-17746-P (Ref. 5).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, every 18 months. This SR is modified by three Notes. Note 1 states that neutron detectors are excluded from the CHANNEL CALIBRATION. The source range neutron detectors are maintained based on manufacturer's recommendations. For the intermediate and power range channels, detector plateau curves are obtained, evaluated, and compared to manufacturer's data. Note 2 states that this test shall include verification that the time constants are adjusted to the prescribed values where applicable. Note 3 states that the power and intermediate range detector plateau voltage verification is not required to be current until 72 hours after achieving equilibrium conditions with THERMAL POWER ≥ 95% RTP. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to perform a meaningful detector plateau voltage verification. The allowance of 72 hours after equilibrium conditions are attained at the testing plateau provides sufficient time to allow power ascension testing to be conducted in a controlled and orderly manner at conditions that provide acceptable results and without introducing the potential for extended operation at high power levels with instrumentation that has not been verified to be OPERABLE for subsequent use.

The 18 month Frequency is based on past operating experience, which has shown these components usually pass the Surveillance when performed on the 18 month Frequency. The conditions for verifying the power and intermediate range detector plateau voltages are described above. The other remaining portions of the CHANNEL CALIBRATIONS may be performed either during a plant outage or during plant operation.

INSERT 3

SR 3.3.1.12

Not Used.

SR 3.3.1.13

SR 3.3.1.13 is the performance of a COT of RTS interlocks every 18 months.

INSERT 3

The Surveillance Requirement is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also states that the methodology for calculating the as-left and the as-found tolerances is specified in WCAP-17746-P (Ref. 5).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.13 (continued)

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of the Manual Reactor Trip, the SI Input from ESFAS, and the Reactor Trip Bypass Breaker undervoltage trip mechanisms. This TADOT is performed every 18 months. The Manual Reactor Trip TADOT shall independently verify the OPERABILITY of the handswitch undervoltage and shunt trip contacts for both the Reactor Trip Breakers and Reactor Trip Bypass Breakers. The Reactor Trip Bypass Breaker test shall include testing of the automatic undervoltage trip mechanism.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.15

SR 3.3.1.15 is the performance of a TADOT of Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to exceeding the P-9 interlock whenever the unit has been in MODE 3. This Surveillance is not required if it has been performed within the previous 31 days. Verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to exceeding the P-9 interlock.

SR 3.3.1.16

SR 3.3.1.16 verifies that the individual channel actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in Table B 3.3.1-~~2~~. No credit was taken in the safety analyses for those

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.16 (continued)

channels with response times listed as N.A. No response time testing requirements apply where N.A. is listed in Table B 3.3.1-~~2~~1. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor until loss of stationary gripper coil voltage.

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time verification is performed with the time constants set at their nominal values. The response time may be measured by a series of overlapping tests, or other verification (e.g., Ref. 7), such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated response times with actual response time tests on the remainder of the channel. Allocations for response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" (Ref. 7), provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

The allocations for sensor response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

As appropriate, each channel's response time must be verified every 18 months on a STAGGERED TEST BASIS. Each verification shall include at least one train such that both trains are verified at least once per 36 months. Testing of the final actuation devices is included in the verification. Response times cannot be determined during unit operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.16 (continued)

surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.3.1.16 is modified by a Note stating that neutron detectors are excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input to the first electronic component in the channel.

REFERENCES

1. USAR, Chapter 7.

2. USAR, Chapter 15.

3. IEEE-279-1971.

4. 10 CFR 50.49.

5. ~~WCNOC Nuclear Safety Analysis Setpoint Methodology for the Reactor Protection System, (TR-89-0001).~~ WCAP-17746-P, Revision 0, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station," [Date TBD].

6. WCAP-10271-P-A and Supplement 1-P-A, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," May 1986.

7. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.

8. WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases," Revision 1, January 1978.

9. IE Information Notice 79-22, "Qualification of Control Systems," September 14, 1979.

10. ~~"Wolf Creek Setpoint Methodology Report," SNP(KG) 492, August 29, 1984.~~ Not used.

11. USAR, Table 15.0-4.

BASES

REFERENCES (continued)

12. WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," October 1998.

13. WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," March 2003.

14. WOG-06-17, "WCAP-10271-P-A Justification for Bypass Test Time and Completion Time Technical Specification Changes for Reactor Trip on Turbine Trip (ITSWG Action item #314)," January 20, 2006.

15. Regulatory Guide 1.105, Revision 3, "Setpoints for Safety Related Instrumentation," December 1999.

16. WCAP-17602-P, Revision 0, "Westinghouse Setpoint Calculations for the Wolf Creek Generating Station Control, Protection, and Indication Systems," [Date TBD].

TABLE B 3.3.1-1
(Page 1 of 2)

FUNCTION	TRIP SETPOINT[a]
1. Manual Reactor Trip	NA
2. Power Range Neutron Flux	
a. High	≤ 109% of RTP
b. Low	≤ 25% of RTP
3. Power Range Neutron Flux	
a. High Positive Rate	≤ 4% of RTP with a time constant ≥ 2 seconds
b. High Negative Rate	≤ 4% of RTP with a time constant ≥ 2 seconds
4. Intermediate Range Neutron Flux	≤ 25% of RTP
5. Source Range Neutron Flux	≤ 10^5 cps
6. Overtemperature ΔT	See Table 3.3.1-1 Note 1
7. Overpower ΔT	See Table 3.3.1-1 Note 2
8. Pressurizer Pressure	
a. Low	≥ 1940 psig
b. High	≤ 2385 psig
9. Pressurizer Water level - High	≤ 92% of instrument span
10. Reactor Coolant Flow - Low	≥ 89.9% of loop design flow (90,324 gpm)
11. Not Used	
12. Undervoltage RCPs	≥ 10578 Vac
13. Underfrequency RCPs	≥ 57.2 Hz
14. Steam Generator (SG) Water Level Low - Low	≥ 23.5% of narrow range instrument span
15. Not Used	
16. Turbine Trip	
a. Low Fluid Oil Pressure	≥ 590.00 psig
b. Turbine Stop Valve Closure	≥ 1% open

TABLE B 3.3.1-1
(Page 2 of 2)

FUNCTION	TRIP SETPOINT[a]
17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	N.A.
18. Reactor Trip System Interlocks	
a. Intermediate Range Neutron Flux, P-6	≥ 1.0E-10 amps
b. Low Power Reactor Trips Block, P-7	N.A.
c. Power Range Neutron Flux, P-8	≤ 48% RTP
d. Power Range Neutron Flux, P-9	≤ 50% RTP
e. Power Range Neutron Flux, P-10	10% RTP
f. Turbine Impulse Pressure, P-13	≤ 10% turbine power
19. Reactor Trip Breakers	N.A.
20. Reactor Trip breaker Undervoltage and Shunt Trip Mechanisms	N.A.
21. Automatic Trip Logic	N.A.

[a] The inequality sign only indicates conservative direction. The as-left value will be within a two-sided calibration tolerance band on either side of the nominal value. This also applies to the Overtemperature ΔT and Overpower ΔT K and τ values.

TABLE B 3.3.1-~~2~~ 1
(Page 1 of 2)

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range Neutron Flux	
a. High	≤ 0.5 second[1]
b. Low	≤ 0.5 second[1]
3. Power Range Neutron Flux	
a. High Positive Rate	≤ 0.5 second[1].
b. High Negative Rate	≤ 0.5 second[1]
4. Intermediate Range Neutron Flux	N.A.
5. Source Range Neutron Flux	N.A.
6. Overtemperature ΔT	≤ 6.0 seconds[1]
7. Overpower ΔT	≤ 6.0 seconds[1]
8. Pressurizer Pressure	
a. Low	≤ 2.0 seconds
b. High	≤ ~~2.0 seconds~~ 1.0 second
9. Pressurizer Water Level - High	N.A.
10. Reactor Coolant Flow - Low	
a. Single Loop (Above P-8)	≤ 1.0 second
b. Two Loops (Above P-7 and below P-8)	≤ 1.0 second
11. Not Used	
12. Undervoltage - Reactor Coolant Pumps	≤ 1.5 seconds
13. Underfrequency - Reactor Coolant Pumps	≤ 0.6 second
14. Steam Generator Water Level - Low-Low	≤ 2.0 seconds
15. Not Used	

[1] Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE B 3.3.1-~~2~~ 1
(Page 2 of 2)

FUNCTIONAL UNIT	RESPONSE TIME
16. Turbine Trip	
a. Low Fluid Oil Pressure	N.A.
b. Turbine Stop Valve Closure	N.A.
17. Safety Injection Input for ESF	N.A.
18. Reactor Trip System Interlocks	N.A.
19. Reactor Trip Breakers	N.A.
20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	N.A.
21. Automatic Trip and Interlock Logic	N.A.

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the ESFAS, as well as specifying LCOs on other reactor system parameters and equipment performance.

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

INSERT 4

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below:

- Field transmitters or process sensors and instrumentation: provide a measurable electronic signal based on the physical characteristics of the parameter being measured;

and Turbine Trip Low Fluid Oil Pressure

- Signal processing equipment including 7300 Process Protection System, and Foxboro Spec 200 (for Auxiliary Feedwater Low Suction Pressure) field contacts, and protection channel sets: provide signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system ~~devices~~, and control board/control room/miscellaneous indications; and

channels

- Solid State Protection System (SSPS) including input, logic, output bays and Balance of Plant (BOP) ESFAS circuitry: initiates the proper unit shutdown or engineered safety feature (ESF) actuation in accordance with the defined logic and based on the bistable outputs from the signal process control and protection system.

Field Transmitters or Sensors

NTSP

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. In many cases, field transmitters or sensors that input to the ESFAS are shared with the Reactor Trip System (RTS). In some cases, the same channels also provide control system inputs. To account for calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the ~~Trip Setpoint and Allowable Values~~. The OPERABILITY of each transmitter or sensor ~~can be evaluated when its "as-found" calibration data are compared against its documented acceptance criteria.~~

is determined by "as-found" calibration data evaluated during the CHANNEL CALIBRATION based on the criteria defined in WCAP-17746-P, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station" (Ref. 6) and WCAP-17602-P, "Westinghouse Setpoint Calculations for the Wolf Creek Generating Station Control, Protection, and Indication Systems," (Ref. 17). The OPERABILITY of each transmitter or sensor may also be determined by qualitative assessment of the field transmitter or sensor as related to the channel behavior observed during performance of the CHANNEL CHECK.

INSERT 4

Technical Specifications are required by 10 CFR 50.36 to include LSSS for variables that have significant safety functions. LSSS are defined by the regulation as "Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a protective action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

The Nominal Trip Setpoint (NTSP) specified in Table 3.3.2-1 is a predetermined setting for a protection channel chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded. As such, the NTSP accounts for uncertainties in setting the channel (e.g., calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the NTSP ensures that SLs are not exceeded. Therefore, the NTSP meets the definition of an LSSS (Ref. 16).

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as "...being capable of performing its safety functions(s)." Relying solely on the NTSP to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protection channel setting during a surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the NTSP due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the NTSP and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protection channel. Therefore, the channel would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the channel within the established as-left tolerance around the NTSP to account for further drift during the next surveillance interval.

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB),
2. Fuel centerline melt shall not occur, and
3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR 50.67 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within the 10 CFR 50.67 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

BASES

Analytical Limits are defined in USAR Chapter 15 (Ref. 3).

BACKGROUND (continued)

Signal Processing Equipment

NTSPs derived from Analytical Limits

the

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with ~~setpoints~~ established by safety analyses. If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the SSPS or BOP ESFAS for decision evaluation. Channel separation is maintained up to and through the input circuitry. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS, while others provide input to the SSPS, BOP ESFAS the main control board, the unit computer, and one or more control systems.

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function is still OPERABLE with a two-out-of-two logic. If one channel fails such that a partial Function trip occurs, a trip will not occur and the Function is still OPERABLE with a one-out-of-two logic.

Generally, if a parameter is used for input to the SSPS and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Again, a single failure will neither cause nor prevent the protection function actuation.

These requirements are described in IEEE-279-1971 (Ref. 4). The actual number of channels required for each unit parameter is specified in Reference 2.

Trip Setpoints ~~and Allowable Values~~

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the two-sided tolerance band for channel accuracy ~~(typically ± 15mv)~~.

BASES

BACKGROUND

Trip Setpoints ~~and Allowable Values~~ (continued)

calculation

the NTSPs specified in Table 3.3.2-1

methodology

WCAP-17746-P, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station," (Ref. 6) and WCAP-17602-P, "Westinghouse Setpoint Calculations for the Wolf Creek Generating Station Control, Protection, and Indication Systems," (Ref. 17).

The Trip Setpoints ~~listed in Table B 3.3.2-1~~ used in the bistables are based on the analytical limits stated in Reference 3. The ~~selection~~ of ~~these Trip Setpoints~~ is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and environment errors for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 5), the ~~Allowable Values~~ specified in Table 3.3.2-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed ~~description of the methodologies~~ used to calculate the ~~Trip Setpoints~~, including their explicit uncertainties, is provided in ~~Reference 10 and was subsequently updated in the~~ "Wolf Creek Nuclear Safety Analysis Setpoint Methodology for the Reactor Protection System" (Ref. 6). The BOP methodology used for Function 6.h is a similar Square Root of the Sum of the Squares methodology as used for the RTS setpoints, except that in the former the instrument span between the Trip Setpoint and the Allowable Value is represented by values for sensor drift, sensor setting tolerance, rack drift, and rack setting tolerance whereas in the latter it is represented only by rack-related terms (rack comparator setting accuracy, rack calibration accuracy, and rack drift). The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a COT. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

NTSPs

The methodology described in Reference 6 incorporates all of the known uncertainties applicable to each channel. The magnitudes of these uncertainties are factored into the determination of each NTSP. Field sensors and signal processing equipment for the associated channels are assumed to operate within the allowances of these uncertainty magnitudes. The as-left and as-found tolerance band methodology is provided in Reference 6. Reference 17 provides the as-left and as-found tolerance band values.

The NTSP is the value at which the bistable is set and is the expected value to be achieved during calibration. The NTSP value is the LSSS and ensures the safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on stated channel uncertainties. Any bistable is considered to be properly adjusted when the as-left NTSP value is within the as-left tolerance band for CHANNEL CALIBRATION uncertainty allowance (i.e., ± rack calibration accuracy). The NTSP value is therefore considered a "nominal" value (i.e., expressed as a value without inequalities) for the purposes of COT and CHANNEL CALIBRATION.

~~Setpoints in accordance with the Allowable Value~~ ensure that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as designed.

NTSPs in conjunction with the use of as-found and as-left tolerances

Each channel can be tested on line to verify that the signal processing equipment and setpoint accuracy is within the specified allowance requirements. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SR section.

~~The Allowable Values listed in Table 3.3.2-1 are based on the methodologies described in Reference 6, which incorporate all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip~~

BASES

BACKGROUND

~~Trip Setpoints and Allowable Values (continued)~~

~~Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.~~

Solid State Protection System

The SSPS equipment is used for the decision logic processing of outputs from the signal processing equipment bistables. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide ESF actuation for the unit. If both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements.

The SSPS performs the decision logic for most ESF equipment actuation; generates the electrical output signals that initiate the required actuation; and provides the status, permissive, and annunciator output signals to the main control room of the unit.

The bistable outputs from the signal processing equipment are sensed by the SSPS equipment and combined into logic matrices that represent combinations indicative of various transients. If a required logic matrix combination is completed, the system will send actuation signals via master and slave relays to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

Each SSPS train has a built in testing device that can automatically test the decision logic matrix functions and the actuation ~~devices~~ channels while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

The actuation of ESF components is accomplished through master and slave relays. The SSPS energizes the master relays appropriate for the condition of the unit. Each master relay then energizes one or more slave relays, which then cause actuation of the end devices. The master and slave relays are routinely tested to ensure operation. The test of the

BASES

BACKGROUND

Solid State Protection System (continued)

master relays energizes the relay, which then operates the contacts and applies a low voltage to the associated slave relays. The low voltage is not sufficient to actuate the slave relays but only demonstrates signal path continuity. The SLAVE RELAY TEST actuates the devices if their operation will not interfere with continued unit operation. For devices that will interfere with continued unit operation, actual component operation is prevented, and slave relay contact operation is verified by a continuity check.

Balance of Plant Engineered Safety Feature Actuation System (BOP ESFAS)

The BOP ESFAS processes signals from SSPS, signal processing equipment and plant radiation monitors to actuate certain ESF equipment. There are two redundant trains of BOP ESFAS, and a third separation group to actuate the turbine driven auxiliary feedwater pump. The redundant trains provide actuation for Auxiliary Feedwater Actuation (motor driven pumps), Containment Purge Isolation, Control Room Emergency Ventilation, and Emergency Exhaust Actutation functions.

The BOP ESFAS has a built-in automatic test insertion (ATI) feature which continuously tests the system logic. Any fault detected during the testing causes an alarm on the main control room overhead annunciator system to alert operators to the problem. Local indications show the test step where the fault was detected.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure - Low is a primary actuation signal for small loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment. Functions such as manual initiation, not specifically credited in the accident safety analysis, are ~~qualitatively~~ credited. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as backups to Functions that were credited in the accident analysis (Ref. 3).

implicitly

INSERT 5

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~The LCO requires all instrumentation performing an ESFAS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.~~

The LCO generally requires OPERABILITY of four or three channels in each instrumentation function and two channels in each logic and manual initiation function. The two-out-of-three and two-out-of-four logic configurations allow one channel to be tripped for maintenance or surveillance testing without causing a reactor trip. In cases where an inoperable channel is placed in the tripped condition indefinitely to satisfy the Required Action of an LCO the logic configurations are reduced to one-out-of-two and one-out-of three where tripping of an additional channel, for any reason, would result in a reactor trip. To allow for surveillance testing or setpoint adjustment of other channels while in this condition, several Required Actions allow the inoperable channel to be bypassed. Bypassing the inoperable channel creates a two-out-of-two or two-out-of-three logic configuration allowing a channel to be tripped for testing without causing a reactor trip. Two logic or manual initiation channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide unit protection in the event of any of the analyzed accidents. ESFAS protection functions are as follows:

1. Safety Injection

   Safety Injection (SI) provides two primary functions:

   1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for limiting peak clad temperature to < 2200°F); and

   2. Boration to ensure recovery and maintenance of SDM (k_{eff} < 1.0).

   These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

INSERT 5

Permissive and interlock setpoints allow the blocking of trips during plant startups, and restoration of trips when the permissive conditions are not satisfied, but they are not explicitly modeled in the safety analyses. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventive or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative starting assumptions for the accident analysis, they are generally considered as nominal values without regard to measurement accuracy.

The LCO requires all instrumentation performing an ESFAS Function, listed in Table 3.3.2-1 in the accompanying LCO, to be OPERABLE. The NTSP specified in Table 3.3.2-1 is OPERABLE if the as-found setpoint is within the as-found tolerance during the CHANNEL CALIBRATION or CHANNEL OPERATIONAL TEST (COT). In this manner, the actual setting of the channel (NTSP) will ensure that a SL is not exceeded at any given point of time as long as the channel has not drifted beyond expected tolerances during the surveillance interval. Note that, although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left tolerance, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned.

If the actual setting of the channel is found to be conservative with respect to the NTSP but is beyond the as-found tolerance band, the channel is OPERABLE but degraded. The degraded condition of the channel will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTSP (within the allowed tolerance) and evaluating the channel response. If the channel is functioning as required and expected to pass the next surveillance, then the channel can be restored to service at the completion of the surveillance.

A trip setpoint may be set more conservative than the NTSP as necessary in response to plant conditions. However, in this case, the OPERABILITY of this instrument must be verified based on the field setting and not the NTSP. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Safety Injection - Automatic Actuation Logic and Actuation Relays (SSPS) (continued)

must be OPERABLE in MODE 4 to support system level manual initiation.

These Functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Unit pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

c. Safety Injection - Containment Pressure - High 1

This signal provides protection against the following accidents:

- SLB inside containment;
- LOCA; and
- Feed line break inside containment.

Containment Pressure - High 1 provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic. The transmitters (d/p cells) and electronics are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment.

NTSP

Thus, the high pressure Function will not experience any adverse environmental conditions and the ~~Trip Setpoint~~ reflects only steady state instrument uncertainties. ~~The Trip Setpoint is ≤ 3.5 psig.~~

BASES

NTSP

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

d. Safety Injection - Pressurizer Pressure - Low (continued)

(LOCA, SLB inside containment, rod ejection). Therefore, the ~~Trip Setpoint~~ reflects the inclusion of both steady state and adverse environment instrument uncertainties. ~~The Trip Setpoint is ≥ 1830 psig.~~

This Function must be OPERABLE in MODES 1, 2, and 3 (above P-11 and below P-11, unless the Safety Injection - Pressurizer Pressure - Low Function is blocked) to mitigate the consequences of an HELB inside containment. This signal may be manually blocked by the operator below the P-11 setpoint. Automatic SI actuation below this pressure setpoint is then performed by the Containment Pressure - High 1 signal.

This Function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. Other ESF functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

e. Safety Injection - Steam Line Pressure - Low

Steam Line Pressure - Low provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG relief or an SG safety valve.

Steam Line Pressure - Low control functions are isolated from the protective functions. Thus, three OPERABLE channels on each steam line are sufficient to satisfy the protective requirements with a two-out-of-three logic on each steam line.

With the transmitters located inside the steam tunnel, it is possible for them to experience adverse environmental conditions during a secondary side break. Therefore, the

NTSP

~~Trip Setpoint~~ reflects both steady state and adverse

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

e. Safety Injection - Steam Line Pressure – Low (continued)

environment instrument uncertainties. ~~The Trip Setpoint is ≥ 615 psig.~~

This Function is anticipatory in nature and has a lead/lag ratio of 50/5.

Steam Line Pressure - Low must be OPERABLE in MODES 1, 2, and 3 (above P-11 and below P-11 unless the Safe Injection - Steam Line Pressure - Low Function is blocked) when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, feed line break is not a concern. Inside containment SLB will be terminated by automatic SI actuation via Containment Pressure - High 1, and outside containment SLB will be terminated by the Steam Line Pressure - Negative Rate - High signal for steam line isolation. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to have a significant effect on required plant equipment.

2. Containment Spray

Containment Spray provides three primary functions:

1. Lowers containment pressure and temperature after an HELB in containment;

2. Reduces the amount of radioactive iodine in the containment atmosphere; and

3. Adjusts the pH of the water in the containment recirculation sumps after a large break LOCA.

These functions are necessary to:

- Ensure the pressure boundary integrity of the containment structure;

- Limit the release of radioactive iodine to the environment; and

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Containment Spray - Automatic Actuation Logic and Actuation Relays (continued)

the primary or secondary systems to pose a threat to containment integrity due to overpressure conditions. Manual initiation is required in MODE 4, even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a containment spray, actuation is simplified by the use of the manual actuation switches. The actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond, to mitigate the consequences of abnormal conditions by manually starting individual components.

c. Containment Spray - Containment Pressure - High 3

This signal provides protection against a LOCA or an SLB inside containment. The transmitters (d/p cells) are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions and the ~~Trip Setpoint~~ NTSP reflects only steady state instrument uncertainties. ~~The Trip Setpoint is ≤ 27.0 psig.~~ Containment Pressure - High 3 feeds the Containment Spray Function and Containment Phase B Isolation Function.

This is one of the few Functions that requires the bistable output to energize to perform its required action. It is not desirable to have a loss of power actuate containment spray, since the consequences of an inadvertent actuation of containment spray could be serious. Note that this Function also has the inoperable channel placed in bypass rather than trip to decrease the probability of an inadvertent actuation.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

(3) Phase B Isolation - Containment Pressure (continued)

The basis for containment pressure MODE applicability and the ~~Trip Setpoint~~ NTSP are as discussed for ESFAS Function 2.c above.

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one SG, at most. For an SLB upstream of the main steam isolation valves (MSIVs), inside or outside of containment, closure of the MSIVs limits the accident to the blowdown from only the affected SG. For an SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the steam lines depressurize. Steam Line Isolation also mitigates the effects of a feed line break and ensures a source of steam for the turbine driven AFW pump during a feed line break.

a. Steam Line Isolation - Manual Initiation

Manual initiation of Steam Line Isolation (fast close) can be accomplished from the control room. There are two push buttons in the control room and either push button can initiate action to immediately close all MSIVs. The LCO requires two channels to be OPERABLE.

b. Steam Line Isolation - Automatic Actuation Logic and Actuation Relays (SSPS)

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

c. Steam Line Isolation – Automatic Actuation Logic (MSFIS)

The LCO requires two trains to be OPERABLE. The Steam Line Isolation signal from SSPS is provided to the Main Steam and Feedwater Isolation System (MSFIS) by four actuation signals per group. The Steam Line Isolation signals are provided by SSPS slave relays K634A and K634B. Actuation logic consists of the circuitry housed within the MSFIS cabinets and extends to the solenoids at the valves responsible for actuating the MSIVs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. Steam Line Isolation (continued)

Manual and automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the RCS and SGs to have an SLB or other accident. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed. In MODES 4, 5, and 6, there is insufficient energy in the

RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

d. Steam Line Isolation - Containment Pressure - High 2

This Function actuates closure of the MSIVs in the event of a LOCA or an SLB inside containment to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. The transmitters (d/p cells) are located outside containment with the sensing line (high pressure side of the transmitter) located inside containment. Containment Pressure - High 2 provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions, and the ~~Trip Setpoint~~ NTSP reflects only steady state instrument uncertainties. ~~The Trip Setpoint is ≤ 17.0 psig.~~

Containment Pressure - High 2 must be OPERABLE in MODES 1, 2, and 3, when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The Steam Line Isolation Function remains OPERABLE in MODES 2 and 3 unless all MSIVs are closed. In MODE 4, the increase in containment pressure following a pipe break would occur over a relatively long time period such that manual actions could reasonably be expected to provide protection. In MODES 5, and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure - High 2 setpoint.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

e. Steam Line Isolation - Steam Line Pressure

(1) Steam Line Pressure - Low

Steam Line Pressure - Low provides closure of the MSIVs in the event of an SLB to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. This Function provides closure of the MSIVs in the event of a feed line break to ensure a supply of steam for the turbine driven AFW pump. Steam Line Pressure - Low was discussed previously under SI Function 1.e~~ and the Trip Setpoint is the same~~.

Steam Line Pressure - Low Function must be OPERABLE in MODES 1, 2, and 3 (above P-11 and below P-11 unless blocked), with any main steam valve open, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. If not blocked below P-11, the Steam Line Pressure - Low Function must be OPERABLE. When blocked, an inside containment SLB will be terminated by automatic actuation via Containment Pressure - High 2. Stuck valve transients and outside containment SLBs will be terminated by the Steam Line Pressure - Negative Rate - High signal for Steam Line Isolation below P-11 when SI has been manually blocked. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have a significant effect on required plant equipment.

(2) Steam Line Pressure - Negative Rate - High

Steam Line Pressure - Negative Rate - High provides closure of the MSIVs for an SLB when less than the P-11 setpoint, to maintain at least one unfaulted SG as a heat sink for the reactor, and to

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

(2) Steam Line Pressure - Negative Rate – High (continued)

limit the mass and energy release to containment. When the operator manually blocks the Steam Line Pressure - Low main steam isolation signal when less than the P-11 setpoint, the Steam Line Pressure - Negative Rate - High signal is automatically enabled. Steam Line Pressure - Negative Rate - High control functions are isolated from the protective functions. Thus, three OPERABLE channels on each steam line are sufficient to satisfy requirements with a two-out-of-three logic.

Steam Line Pressure - Negative Rate - High must be OPERABLE in MODE 3 when the Steam Line Pressure - Low signal is blocked, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3, when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure - Low signal is automatically enabled. The Steam Line Isolation Function is required to be OPERABLE in MODE 3 unless all MSIVs are closed. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to have an SLB or other accident that would result in a release of significant enough quantities of energy to cause a cooldown of the RCS.

NTSP

While the transmitters may experience elevated ambient temperatures due to an SLB, the trip function is based on rate of change, not the absolute accuracy of the indicated steam pressure. Therefore, the ~~Trip Setpoint~~ reflects only steady state instrument uncertainties. ~~The Trip Setpoint is ≤ 100 psi with a rate /lag controller time constant ≥ 50 seconds.~~

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

c. Turbine Trip and Feedwater Isolation - Steam Generator Water Level - High High (P-14)

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments provide input to the SG Water Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with a two-out-of-four logic.

NTSP

The transmitters (d/p cells) are located inside containment. However, the events that this Function protects against cannot cause a severe environment in containment. Therefore, the ~~Trip Setpoint~~ reflects only steady state instrument uncertainties. ~~The Trip Setpoint is ≤ 78% of narrow range span.~~

d. Turbine Trip and Feedwater Isolation - Safety Injection

Turbine Trip and Feedwater Isolation are also initiated by all Functions that initiate SI. The Feedwater Isolation Function requirements for these initiation Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

Turbine Trip and Feedwater Isolation Function 5.c, SG Water Level - High High must be OPERABLE in MODES 1 and 2 except when all MFIVs are closed and de-activated; and all MFRVs are closed and de-activated or closed and isolated by a closed manual valve; and all MFRV bypass valves are closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves. In MODES 3, 4, 5, and 6, Function 5.c is not required to be OPERABLE. The Automatic Actuation Logic and Actuation Relays (SSPS) Function must be OPERABLE in MODE 1, MODE 2 (except when all MFIVs are closed and de-activated; and all MFRVs are closed and de-activated or closed and isolated by a closed manual valve; and all MFRV bypass valves are closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves) and

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

d. Auxiliary Feedwater – Steam Generator Water level – Low Low

SG Water Level - Low Low provides protection against a loss of heat sink. A feed line break, inside or outside of containment, or a loss of MFW, would result in a loss of SG water level. SG Water Level - Low Low provides input to the SG Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system which may then require a protection function actuation and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with two-out-of-four logic.

NTSP

With the transmitters (d/p cells) located inside containment and thus possibly experiencing adverse environmental conditions (feed line break), the ~~Trip Setpoint~~ reflects the inclusion of both steady state and adverse environment instrument uncertainties. ~~The Trip Setpoint for the Start Motor-Driven Pumps and the Start Turbine-Driven Pumps is ≥ 23.5% of narrow range instrumentation span.~~

e. Auxiliary Feedwater - Safety Injection

An SI signal also starts the motor driven AFW pumps via the LOCA sequencer. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Function 1, SI, is referenced for all initiating functions and requirements.

f. Auxiliary Feedwater - Loss of Offsite Power

A loss of offsite power (LOP) to the safeguard buses will be accompanied by a loss of reactor coolant pumping power and the subsequent need for some method of decay heat removal. The loss of offsite power (LOP) is detected by a voltage drop on each safeguard bus. The LOP is sensed and processed by the circuitry for LOP DG start (load shedder and emergency load sequencer) and fed to the BOP ESFAS by the relay actuation. Loss of power to either safeguard bus will start the turbine driven AFW pump to ensure that at least one SG contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip and automatically isolate the SG blowdown and sample lines.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Auxiliary Feedwater (continued)

SR 3.3.2.8. This limits the potential for inadvertent AFW actuations during normal startups and shutdowns. In MODES 3, 4, and 5, the MFW pumps may be normally shut down, and thus pump trip is not indicative of a condition requiring automatic AFW initiation.

h. Auxiliary Feedwater - Pump Suction Transfer on Suction Pressure – Low

transmitters

A low pressure signal in the AFW pump suction line protects the AFW pumps against a loss of the normal supply of water for the pumps, the CST. Three ~~pressure switches~~ are located on the AFW pump suction line from ~~the~~ CST. A low pressure signal sensed by any two of the three ~~switches~~ coincident with an auxiliary feedwater actuation signal will cause the emergency supply of water for both pumps to be aligned. ESW (safety grade) is automatically lined up to supply the AFW pumps to ensure an adequate supply of water for the AFW System to maintain at least one of the SGs as the heat sink for reactor decay heat and sensible heat removal.

NTSP

Since the ~~detectors~~ are located in an area not affected by HELBs or high radiation, they will not experience any adverse environmental conditions and the ~~Trip Setpoint~~ reflects only steady state instrument uncertainties. ~~The Trip Setpoint is ≥ 21.60 psia.~~

This Function must be OPERABLE in MODES 1, 2, and 3 to ensure a safety grade supply of water for the AFW System to maintain the SGs as the heat sink for the reactor. This Function does not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW automatic suction transfer does not need to be OPERABLE because RHR will already be in operation, or sufficient time is available to place RHR in operation, to remove decay heat.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7. Automatic Switchover to Containment Sump

At the end of the injection phase of a LOCA, the RWST will be empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the RHR pumps is automatically switched to the containment recirculation sumps. The low head residual heat removal (RHR) pumps and containment spray pumps draw the water from the containment recirculation sumps, the RHR pumps pump the water through the RHR heat exchanger, inject the water back into the RCS, and supply the cooled water to the other ECCS pumps. Switchover from the RWST to the containment sumps must occur before the RWST empties to prevent damage to the RHR pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sumps to support ECCS pump suction.

a. Automatic Switchover to Containment Sump - Automatic Actuation Logic and Actuation Relays (SSPS)

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Automatic Switchover to Containment Sump - Refueling Water Storage Tank (RWST) Level - Low Low Coincident With Safety Injection

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. A low low-1 level in the RWST coincident with an SI signal provides protection against a loss of water for the ECCS pumps and indicates the end of the injection phase of the LOCA. The RWST is equipped with four level transmitters. These transmitters provide no control functions. Therefore, a two-out-of-four logic is adequate to initiate the protection function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability.

NTSP

The RWST Low Low-1 ~~Allowable Value/Trip Setpoint~~ is selected to ensure switchover occurs before the RWST empties, to prevent ECCS pump damage.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Automatic Switchover to Containment Sump - Refueling Water Storage Tank (RWST) Level - Low Low Coincident With Safety Injection (continued)

NTSP

The transmitters are located in an area not affected by HELBs or post accident high radiation. Thus, they will not experience any adverse environmental conditions and the ~~Trip Setpoint~~ reflects only steady state instrument uncertainties. ~~The Trip Setpoint is ≥ 36% of span.~~

Automatic switchover occurs only if the RWST low low-1 level signal is coincident with SI. This prevents accidental switchover during normal operation. Accidental switchover could damage ECCS pumps if they are attempting to take suction from an empty sump. This is one of the few functions that requires the bistable output to energize to perform its required action. The automatic switchover Function requires the SI Function for OPERABILITY. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

These Functions must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for a LOCA to occur, to ensure a continued supply of water for the ECCS pumps. These Functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. System pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

8. Engineered Safety Feature Actuation System Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlock Functions back up manual actions to ensure bypassable functions are in operation under the conditions assumed in the safety analyses.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Engineered Safety Feature Actuation System Interlocks - Reactor Trip, P-4 (continued)

The turbine trip function, MFW isolation coincident with low T_{avg} function, arming of the steam dump valves function do not serve a mitigation function in the licensing basis safety analyses. Block of the SI signals is required to support long-term ECCS operation in the post-LOCA recirculation mode. Block of the opening of the MFW isolation valves on SI or SG Water Level – High High prevents reopening the valves for mitigation of a high water level in the SGs, which could result in carryover of water into the steam lines and excessive cooldown of the primary system.

The RTB position switches that provide input to the P-4 interlock (P-4 generated when one train's RTB and the alternate train's Bypass Breaker are both open) only function to energize or de-energize or open or close contacts. Therefore, this Function has no adjustable trip setpoint with which to associate a ~~Trip Setpoint and Allowable Value~~. NTSP

This Function does not have to be OPERABLE in MODE 4, 5, or 6 because the main turbine, the MFW System, and the Steam Dump System are not in operation.

b. Engineered Safety Feature Actuation System Interlocks - Pressurizer Pressure, P-11

The P-11 interlock permits a normal unit cooldown and depressurization without actuation of SI or main steam line isolation. With two-out-of-three pressurizer pressure channels (discussed previously) less than the P-11 setpoint, the operator can manually block the Pressurizer Pressure - Low and Steam Line Pressure - Low SI signals and the Steam Line Pressure - Low steam line isolation signal (previously discussed). When the Steam Line Pressure - Low steam line isolation signal is manually blocked, a main steam isolation signal on Steam Line Pressure - Negative Rate - High is automatically enabled. This provides protection for an SLB by closure of the MSIVs. With two-out-of-three pressurizer pressure channels above the P-11 setpoint, the Pressurizer Pressure - Low and Steam Line Pressure - Low SI signals and the Steam Line Pressure - Low steam line isolation signal are automatically enabled. The operator can also

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Engineered Safety Feature Actuation System Interlocks - Pressurizer Pressure, P-11 (continued)

NTSP

enable these trips by use of the respective manual unblock (reset) buttons. When the Steam Line Pressure - Low steam line isolation signal is enabled, the main steam isolation on Steam Line Pressure - Negative Rate - High is disabled. The ~~Trip Setpoint~~ reflects only steady state instrument uncertainties. ~~The Trip Setpoint is ≤ 1970 psig.~~

This Function must be OPERABLE in MODES 1, 2, and 3 to allow an orderly cooldown and depressurization of the unit without the actuation of SI or main steam isolation. This Function does not have to be OPERABLE in MODE 4, 5, or 6 because system pressure must already be below the P-11 setpoint for the requirements of the heatup and cooldown curves to be met.

The ESFAS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

NTSP cannot be re-set to within the two-sided as-left tolerance band or the channel is not functioning as required

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1.

In the event a channel's ~~Trip Setpoint is found nonconservative with respect to the Allowable Value~~, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s)entered for the protection Function(s) affected. When the Required Channels in Table 3.3.2-1 are specified on a per steam line, per SG, per pump, etc., basis, then the Condition may be entered separately for each steam line, SG, pump, etc., as appropriate.

When the number of inoperable channels in a trip function exceed those specified in one or other related Conditions associated with a trip function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies to all ESFAS protection functions.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.4 (continued)

large enough to demonstrate signal path continuity. This test is performed every 92 days on a STAGGERED TEST BASIS. The time allowed for the testing (4 hours) is justified in Reference 7. The Frequency of every 92 days on a STAGGERED TEST BASIS is justified in Reference 13.

The difference between the current "as-found" values and the previous test "as-left" values must be consistent with the drift allowance used in the setpoint methodology (Reference 6). The setpoint shall be left set consistent with the assumptions of the setpoint methodology described in Reference 6. The "as-found" and "as-left" values must also be recorded and trended for consistency with the assumptions of References 6 and 17.

SR 3.3.2.5

SR 3.3.2.5 is the performance of a COT.

A COT is performed on each required channel to ensure the channel will perform the intended Function. ~~Setpoints must be found within the Allowable Values specified in Table 3.3.2-1.~~

~~The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.~~

INSERT 6

The Frequency of 184 days is justified in Reference 13.

SR 3.3.2.6

SR 3.3.2.6 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE is either allowed to function, or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation MODE is prevented from operation by the slave relay blocking circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 18 months. This Frequency is based on relay reliability assessments presented in WCAP-13878-P-A, "Reliability Assessment of Potter & Brumfield MDR Series Relays," (Ref. 9). The reliability assessments are relay specific and apply only to Potter & Brumfield MDR series relays.

For Function 4.c (Steam Line Isolation – Automatic Actuation Logic (MSFIS)) and Function 5.b (Turbine Trip and Feedwater Isolation – Automatic Actuation Logic (MSFIS)), SR 3.3.2.6 is performed on the associated slave relays in the SSPS cabinets and includes verification that the slave relays are energized at the MSFIS cabinets.

INSERT 6

The Surveillance Requirement is modified by two Notes as identified in Table 3.3.2-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also states that the methodology for calculating the as-left and the as-found tolerances is specified in WCAP-17746-P (Ref. 6).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.7

SR 3.3.2.7 is the performance of a TADOT every 18 months. This test is a check of the Loss of Offsite Power function. The trip actuating devices tested within the scope of SR 3.3.2.7 are the LSELS output relays and BOP ESFAS separation groups logic associated with the auto-start of the turbine driven AFW pump upon an ESF bus undervoltage condition.

The SR is modified by a Note that excludes verification of setpoints for relays. The Frequency is adequate. It is based on industry operating experience, considering instrument reliability and operating history data and is consistent with the typical refueling cycle. The trip actuating devices tested have no associated setpoint.

SR 3.3.2.8

SR 3.3.2.8 is the performance of a TADOT. This test is a check of the Manual Actuation Functions (SSPS) and AFW pump start on trip of all MFW pumps BOP ESFAS. The Manual Safety Injection TADOT shall independently verify OPERABILITY of the handswitch undervoltage and shunt trip contacts for both the Reactor Trip Breakers and Reactor Trip Bypass Breakers as well as the contacts for safety injection actuation. It is performed every 18 months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.). The Frequency is adequate, based on industry operating experience and is consistent with the typical refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT for manual initiation Functions. The manual initiation Functions have no associated setpoints.

SR 3.3.2.9

SR 3.3.2.9 is the performance of a CHANNEL CALIBRATION.

The difference between the current "as-found" values and the previous test "as-left" values when trended must be consistent with the drift allowance used in the setpoint methodology (Reference 6).

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.9 (continued)

The Frequency of 18 months is based on the assumed calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

This SR is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable. This does not include verification of time delay relays. These are verified by response time testing per SR 3.3.2.10.

INSERT 6

SR 3.3.2.10

1

This SR verifies the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response Time verification acceptance criteria are included in Table B 3.3.2-~~2~~. Table B 3.3.2-~~2~~ format is based on the initiating trip signal. No credit was taken in the safety analyses for those channels with response times listed as N.A. No response time testing requirements apply where N.A. is listed in Table B 3.3.2-~~2~~. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., pumps at rated discharge pressure, valves in full open or closed position).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time verification is performed with the time constants set at their nominal values. The response time may be verified by a series of overlapping tests, or other verification (e.g., Ref. 8), such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" (Ref. 7), provides the basis and methodology for using allocated sensor

INSERT 6

The Surveillance Requirement is modified by two Notes as identified in Table 3.3.2-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also states that the methodology for calculating the as-left and the as-found tolerances is specified in WCAP-17746-P (Ref. 6).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.10 (continued)

response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

The allocations for sensor response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

The NRC approved the use of ASME Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light-Water Reactor Plants," as an alternative to stroke time testing for motor-operated valves (Ref. 14). The parameters that must be present to achieve the analyzed response time under design basis conditions are measured to ensure the valve is capable of performing its safety function. This process verifies design basis capability, including response time, and is a significant improvement over simple stroke time measurement. This process allows the establishment of periodic valve test intervals if there is assurance that the valve will remain capable of performing its safety function throughout the interval.

ESF response times specified in Table B 3.3.2-~~2~~ 1 which include sequential operation of RWST and VCT valves (Notes 3 and 4) are based on values assumed in the non-LOCA safety analyses. These analyses take credit for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When the sequential operation of the RWST and VCT valves is not included in the response times (Note 7), the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times specified in Table B 3.3.2-~~2~~ 1 will assure that the assumptions used for the LOCA and non-LOCA analyses with respect to the operation of the VCT and RWST valves are valid.

ESF RESPONSE TIME verification is performed on an 18 month STAGGERED TEST BASIS. Each verification shall include at least one train such that both trains are verified at least once per 36 months. Testing of the final actuation devices, which make up the bulk of the

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.10 (continued)

response time, is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every 18 months. The 18 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

This SR is modified by a Note that clarifies that the turbine driven AFW pump is tested within 24 hours after reaching 900 psig in the SGs.

SR 3.3.2.11

SR 3.3.2.11 is the performance of a TADOT as described in SR 3.3.2.8, except that it is performed for the P-4 Reactor Trip Interlock, and the Frequency is every 18 months. This Frequency is based on operating experience.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no associated setpoint. This TADOT does not include the circuitry associated with steam dump operation since it is control grade circuitry.

The difference between the current "as-found" values and the previous test "as-left" values must be consistent with the drift allowance used in the setpoint methodology (Reference 6). The setpoint shall be left set consistent with the assumptions of the setpoint methodology described in Reference 6. The "as-found" and "as-left" values must also be recorded and trended for consistency with the assumptions of References 6 and 17.

SR 3.3.2.12

SR 3.3.2.12 is the performance of a monthly COT on ESFAS Function 6.h, "Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low."

A COT is performed to ensure the channel will perform the intended Function. ~~Setpoints must be found within the Allowable Values specified in Table 3.3.2-1.~~

~~The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.~~

INSERT 6

INSERT 6

The Surveillance Requirement is modified by two Notes as identified in Table 3.3.2-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also states that the methodology for calculating the as-left and the as-found tolerances is specified in WCAP-17746-P (Ref. 6).

BASES

REFERENCES

1. USAR, Chapter 6.
2. USAR, Chapter 7.
3. USAR, Chapter 15.
4. IEEE-279-1971.
5. 10 CFR 50.49.
6. ~~WCNOC Nuclear Safety Analysis Setpoint Methodology for the Reactor Protection System, TR 89-0001~~.

   WCAP-17746-P, Revision 0, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station," [Date TBD].
7. WCAP-10271-P-A Supplement 2, Revision 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System," June 1990.
8. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.
9. WCAP-13878-P-A, Revision 2, "Reliability Assessment of Potter & Brumfield MDR Series Relays," August 2000.
10. ~~"Wolf Creek Setpoint Methodology Report," SNP (KG) 492, August 29, 1984.~~

    Not used.
11. Amendment No. 43 to Facility Operating License No. NPF-42, March 29, 1991.
12. WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," October 1998.
13. WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," March 2003.
14. 10 CFR 50.55a(b)(3)(iii), Code Case OMN-1.
15. Performance Improvement Request (PIR) 2005-2067.
16. Regulatory Guide 1.105, Revision 3, "Setpoints for Safety Related Instrumentation," December 1999.
17. WCAP-17602-P, Revision 0, "Westinghouse Setpoint Calculations for the Wolf Creek Generating Station Control, Protection, and Indication Systems," [Date TBD].

TABLE B 3.3.2-1
(Page 1 of 2)

FUNCTION	TRIP SETPOINT[a]
1. Safety Injection	
a. Manual Initiation	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
c. Containment Pressure – High-1	≤ 3.5 psig
d. Pressurizer Pressure - Low	≥ 1830 psig
e. Steam Line Pressure - Low	≥ 615 psig
2. Containment Spray	
a. Manual Initiation	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
c. Containment Pressure - High-3	≤ 27.0 psig
3. Containment Isolation	
a. Phase A Isolation	
(1) Manual Initiation	N.A.
(2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
(3) Safety Injection	See Function 1 (Safety Injection)
b. Phase B Isolation	
(1) Manual Initiation	N.A.
(2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
(3) Containment Pressure - High-3	≤ 27.0 psig
4. Steam Line Isolation	
a. Manual Initiation	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
c. Automatic Actuation Logic (MSFIS)	N.A.
d. Containment Pressure - High-2	≤ 17.0 psig
e. Steam Line Pressure	
(1) Low	≥ 615 psig
(2) Negative Rate - High	≤ 100 psi

TABLE B 3.3.2-1
(Page 2 of 2)

FUNCTION	TRIP SETPOINT[a]
5. Turbine Trip and Feedwater Isolation	
a. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
b. Automatic Actuation Logic (MSFIS)	N.A.
c. SG Water Level - High High	≤ 78% of narrow range instrument span
d. Safety Injection	See Function 1 (Safety Injection)
6. Auxiliary Feedwater	
a. Manual Initiation	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.
d. SG Water Level - Low-Low	≥ 23.5% of narrow range instrument span
e. Safety Injection	See Function 1 (Safety Injection)
f. Loss of Offsite Power	N.A.
g. Trip of all Main Feedwater Pumps	N.A.
h. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	≥ 21.60 psia
7. Automatic Switchover to Containment Sump	
a. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
b. Refueling Water Storage Tank (RWST) Level - Low Low	≥ 36% of instrument span
Coincident with Safety Injection	See Function 1 (Safety Injection)
8. ESFAS Interlocks	
a. Reactor Trip, P-4	N.A.
b. Pressurizer Pressure, P-11	≤ 1970 psig

[a] The inequality sign only indicates conservative direction. The as-left value will be within a two-sided calibration tolerance band on either side of the nominal value.

Table B 3.3.2-~~2~~ 1
(Page 1 of 3)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Containment Purge Isolation	N.A.
f. Steam Line Isolation	N.A.
g. Feedwater Isolation	N.A.
h. Auxiliary Feedwater	N.A.
I. Essential Service Water	N.A.
j Containment Cooling	N.A.
k. Control Room Isolation	N.A.
l. Reactor Trip	N.A.
m. Emergency Diesel Generators	N.A.
n. Component Cooling Water	N.A.
o. Turbine Trip	N.A.
2. Containment Pressure - High-1	
a. Safety Injection (ECCS)	$\leq 29^{(7)}/27^{(4)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 2^{(5)}$
3) Phase "A" Isolation	$\leq 1.5^{(5)}$
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	$\leq 60^{(1)}$
6) Containment Cooling	$\leq 60^{(1)}$
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	$\leq 14^{(6)}$
9) Turbine Trip	N.A.
3. Pressurizer Pressure - Low	
a. Safety Injection (ECCS)	$\leq 29^{(7)}/27^{(4)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 2^{(5)}$
3) Phase "A" Isolation	$\leq 2^{(5)}$
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	$\leq 60^{(1)}$
6) Containment Cooling	$\leq 60^{(1)}$
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	$\leq 14^{(6)}$
9) Turbine Trip	N.A.

Table B 3.3.2-~~2~~ 1
(Page 2 of 3)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Steam Line Pressure - Low	
a. Safety Injection (ECCS)	≤ 39[3]/27[4]
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 2[5]
3) Phase "A" Isolation	≤ 2[5]
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	≤ 60[1]
6) Containment Cooling	≤ 60[1]
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	≤ 14[6]
9) Turbine Trip	N.A.
b. Steam Line Isolation	≤ 2[5]
5. Containment Pressure - High-3	
a. Containment Spray	≤ 32[1]/20[2]
b. Phase "B" Isolation	≤ 31.5
6. Containment Pressure - High-2	
Steam Line Isolation	≤ 2[5]
7. Steam Line Pressure - Negative Rate-High	
Steam Line Isolation	≤ 2[5]
8. Steam Generator Water Level - High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 2[5]
9. Steam Generator Water Level - Low-Low	
a. Start Motor Driven Auxiliary Feedwater Pumps	≤ 60
b. Start Turbine Driven Auxiliary Feedwater Pumps	≤ 60
10. Loss-of-Offsite Power	
Start Turbine Driven Auxiliary Feedwater Pumps	≤ 60
11. Trip of All Main Feedwater Pumps	
Start Motor Driven Auxiliary Feedwater Pumps	N.A.

Table B 3.3.2-~~2~~ 1
(Page 3 of 3)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
12. Auxiliary Feedwater Pump Suction Pressure-Low Transfer to Essential Service Water	≤ 60[1]
13. RWST Level-Low-Low Coincident with Safety Injection Automatic Switchover to Containment Sump	≤ 60

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting delay not included. Offsite power available.

(3) Diesel generator starting and sequence loading delay included. RHR pumps not included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(4) Diesel generator starting and sequence loading delays not included. Offsite power available. RHR pumps not included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(5) Does not include valve closure time.

(6) Includes time for diesel to reach full speed.

(7) Diesel generator starting and sequence loading delays included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is not included. Response time assumes only opening of RWST valves.

BASES

BACKGROUND (continued)

RCP. There are four of these 8-second timers per bus, one for each degraded voltage channel. The bistable outputs are then combined in a two-out-of-four logic to generate a degraded voltage signal if the voltage is below approximately 90%. Once the two-out-of four logic is satisfied, contacts in the bus feeder breaker trip circuits closed to arm the tripping circuitry. If a safety injection signal (SIS) were to occur concurrently with or after the arming of the tripping circuitry, the bus feeder breaker would open immediately, a bus undervoltage would be sensed, and a LOP signal would be generated. Should the degraded voltage condition occur in a non-accident condition (no SIS present), an additional 111 second time delay is provided. These time delays are specific to the feeder breakers (2 per bus). If the degraded voltage is not alleviated in the overall 119 seconds (nominal delay), the bus feeder breaker is tripped.

Each Nominal Trip Setpoint (NTSP) specified is more conservative than the analytical limit assumed in the operational or analytical limits analyses in order to account for instrument uncertainties appropriate to the trip function.

OPERABILITY of LSELS is addressed in LCO 3.8.1, "AC Sources - Operating," And LCO 3.8.2, "AC Sources - Shutdown."

Trip Setpoints ~~and Allowable Values~~ /bistables

3 4

The Trip Setpoints used in the relays are based on References ~~1~~ and ~~2~~. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account.

relays/bistables

~~The actual nominal Trip Setpoint entered into the relays is normally still more conservative than that required by the Allowable Value.~~ The Trip Setpoints are the nominal value at which the ~~bistables~~ are set. Any bistable is considered to be properly adjusted when the "as left" value is within the two-sided tolerance band for channel accuracy. If the measured setpoint ~~does not exceed the Allowable Value~~, the relay is considered OPERABLE.

relay/ /bistable

can be reset to within the two-sided as-left tolerance band

The as-left and as-found tolerance band methodology is provided in Reference 3. Reference 4 provides the as-left and as-found tolerance band values.

Setpoints adjusted in accordance with the ~~Allowable Value~~ ensure that the consequences of accidents will be acceptable, provided the unit is operated from within the LCOs at the onset of the accident and that the equipment functions as designed.

two-sided as-left tolerance band

NTSPs

~~Allowable Values and/or nominal Trip Setpoints~~ are specified for each Function in SR 3.3.5.3. ~~Nominal Trip Setpoints~~ are also specified in the unit specific setpoint calculations. The ~~nominal setpoints~~ are selected to ensure that the setpoint measured by the surveillance procedure does not exceed the ~~Allowable Value~~ if the relay is performing as required. If the

NTSPs

operational or analytical limits /bistable

A Trip Setpoint less conservative than the NTSP (field setting) is acceptable provided that operation and the as-left and as-found tolerances are consistent with the assumptions of the setpoint calculation.

two-sided tolerance band

/bistable

BASES

BACKGROUND

Trip Setpoints ~~and Allowable Values~~ (continued)

measured setpoint does not exceed the ~~Allowable Value~~, the relay is considered OPERABLE. ~~Operation with a Trip Setpoint less conservative than the~~ nominal Trip Setpoint, but within the Allowable Value, is acceptable ~~provided~~ that operation and testing is ~~consistent~~ with the assumptions of the setpoint ~~calculation. Each~~ Allowable Value and/or Trip Setpoint specified is more ~~conservative than the~~ analytical limit assumed in the transient ~~and~~ accident analyses in order to ~~account~~ for instrument ~~uncertainties~~ appropriate to the trip function.

APPLICABLE SAFETY ANALYSES

The LOP DG start instrumentation is required for the Engineered Safety Features (ESF) Systems to function in any accident with a loss of offsite power. Its design basis is that of the ESF Actuation System (ESFAS).

Accident analyses credit the loading of the DG based on the loss of offsite power during a loss of coolant accident (LOCA). The actual DG start has historically been associated with the ESFAS actuation. The DG loading has been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power. The analyses assume a non-mechanistic DG loading, which does not explicitly account for each individual component of loss of power detection and subsequent actions.

The required channels of LOP DG start instrumentation, in conjunction with the ESF systems powered from the DGs, provide unit protection in the event of any of the analyzed accidents discussed in Reference 2, in which a loss of offsite power is assumed.

The delay times assumed in the safety analysis for the ESF equipment include the 12 second DG start delay, and the appropriate sequencing delay, if applicable. The response times for ESFAS actuated equipment in Bases Table B 3.3.2-~~2~~ include the appropriate DG loading and sequencing delay.

1

The LOP DG start instrumentation channels satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO for LOP DG start instrumentation requires that four channels per 4.16 kV NB system bus of both the loss of voltage and degraded voltage Functions shall be OPERABLE in MODES 1, 2, 3, and 4 when the LOP DG start instrumentation supports safety systems associated with the ESFAS. In MODES 5 and 6, the four channels must be OPERABLE

BASES

LCO (continued)

whenever the associated DG is required to be OPERABLE to ensure that the automatic start of the DG is available when needed. Loss of the LOP DG Start Instrumentation Function could result in the delay of safety systems initiation when required. This could lead to unacceptable consequences during accidents. During the loss of offsite power the DG powers the motor driven auxiliary feedwater pumps which are automatically started after expiration of the appropriate time delays by the load shedder and emergency load sequencer. Failure of these pumps to start would leave the turbine driven pump, started by the BOP ESFAS directly upon receipt of a loss of voltage signal from the load shedder emergency and load sequencer output relays as well as an increased potential for a loss of decay heat removal through the secondary system. OPERABILITY of the load shedder and emergency load sequencer is addressed in LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown."

APPLICABILITY

The LOP DG Start Instrumentation Functions are required in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation in MODE 5 or 6 is required whenever the required DG must be OPERABLE so that it can perform its function on an LOP or degraded power to the vital bus.

NTSP cannot be re-set to within the two-sided as-left tolerance band

ACTIONS

In the event a channel's ~~Trip Setpoint is found nonconservative with respect to the Allowable Value,~~ or the channel is found inoperable, then the function that channel provides must be declared inoperable and the LCO Condition entered for the particular protection function affected.

Because the required channels are specified on a per bus basis, the Condition may be entered separately for each bus as appropriate.

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in the LCO. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the LOP DG start Function with one loss of voltage or one degraded voltage channel per bus inoperable.

BASES

ACTIONS

B.1 (continued)

MODES 1 - 4 and takes into account the low probability of an event requiring an LOP start occurring during this interval. When the associated DG is required to be OPERABLE in MODES 5 and 6, the Completion Time of Required Action C.1 in LCO 3.8.2, "AC Sources - Shutdown," is consistent with the required times for actions requiring prompt action.

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1

Not Used.

SR 3.3.5.2

SR 3.3.5.2 is the performance of a TADOT. This test is performed every 31 days. The test checks trip devices that provide actuation signals directly, bypassing the analog process control equipment. For these tests, the relay Trip Setpoints are verified and adjusted as necessary. The SR is modified by a Note that excludes verification of time delays. Testing of the time delay relays is performed as part of the CHANNEL CALIBRATION (SR 3.3.5.3). The Frequency is based on the known reliability of the relays and controls and the multichannel redundancy available, and has been shown to be acceptable through operating experience. If the measured setpoint does not exceed the ~~Allowable Value~~, the trip device is considered OPERABLE.

two-sided as-left tolerance band

There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

SR 3.3.5.3

SR 3.3.5.3 is the performance of a CHANNEL CALIBRATION.

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

~~Calculation XX-E-009 (Ref. 3) calculates the undervoltage/degraded voltage setpoints for the NB/NG relays. The calculation also ensures adequate voltage will be present at the end use loads under minimum switchyard voltage and maximum accident loading. Calculation XX-E-009 identifies that the minimum acceptable voltage for the NB01 bus is 3707 V (105.9 V after PT) and for the NB02 bus is 3704 V (105.9 V after PT).~~

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.3 (continued)

The Frequency of 18 months is based on operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.5.4

SR 3.3.5.4 is the performance of the required response time verification every 18 months on a STAGGERED TEST BASIS. This SR measures the total response time of the undervoltage relays, logic circuitry and EDG start time. Response time verification acceptance criteria are:

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME
Loss of Power	
a. 4kV Bus Undervoltage - Loss of Voltage	≤ 14 seconds
b. 4kV Bus Undervoltage - Grid Degraded Voltage	≤ 144 seconds

Each verification shall include at least one train such that both trains are verified at least once per 36 months.

REFERENCES

1. USAR, Section 8.3.
2. USAR, Chapter 15.
3. ~~Calculation XX-E-009, "System NB, NG, PG Undervoltage/Degraded Voltage Relay Setpoints."~~

WCAP-17746-P, Revision 0, "Westinghouse Setpoint Methodology as Applied to the Wolf Creek Generating Station," [Date TBD].

4. WCAP-17602-P, Revision 0, "Westinghouse Setpoint Calculations for the Wolf Creek Generating Station Control, Protection, and Indication Systems," [Date TBD].

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.2 (continued)

STAGGERED TEST BASIS. The Frequency is adequate based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.6.3

A COT is performed every 92 days on each required channel to ensure the entire channel will perform the intended Function. The Frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 2). This test verifies the capability of the instrumentation to provide the containment purge system isolation. The trip setpoint concentration value (uCi/cm^3) is to be established such that the actual submersion rate would not exceed 9 mr/h in the containment building. The setpoint value may be increased up to the equivalent limits of Section 3.1 of the ODCM in accordance with the methodology and parameters in the ODCM during containment purge or vent provided the setpoint value does not exceed twice the maximum concentration activity in the containment determined by the sample analysis performed prior to each release in accordance with Table 3-1 of the ODCM.

There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

SR 3.3.6.4

SR 3.3.6.4 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and is performed every 18 months. Each Manual Actuation Function is tested through the BOP ESFAS logic.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.5

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

SR 3.3.6.6

SR 3.3.6.6 is the performance of the required response time verification every 18 months on a STAGGERED TEST BASIS. Response time verification acceptance criteria for the containment purge isolation instrumentation is ≤ 2 seconds. This response time acceptance criteria does not include valve closure time. Each verification shall include at least one train such that both trains are verified at least once per 36 months.

REFERENCES

1. 10 CFR 100.11.

2. NUREG-1366, July 22, 1993.

No Changes to This Page.
Included for Information Only

BASES

ACTIONS (continued)

E.1 and E.2

Condition E applies when the Required Action and associated Completion Time for Conditions A, B or C have not been met when irradiated fuel assemblies are being moved. Movement of irradiated fuel assemblies and CORE ALTERATIONS must be suspended immediately to reduce the risk of accidents that would require CREVS actuation. This does not preclude movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.7-1 determines which SRs apply to which CREVS Actuation Functions.

SR 3.3.7.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.7.2

A COT is performed once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.7.2 (continued)

verifies the capability of the instrumentation to provide the CREVS actuation. The setpoints shall be left consistent with Note (b) of Table 3.3.7-1. The Frequency is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.7.3

SR 3.3.7.3 is the performance of an ACTUATION LOGIC TEST using the BOP ESFAS automatic tester. The continuity check does not have to be performed, as explained in the Note. This SR is applied to the balance of plant actuation logic and relays that do not have circuits installed to perform the continuity check. This test is required every 31 days on a STAGGERED TEST BASIS. The Frequency is adequate based on industry operating experience, considering instrument reliability and operating history data.

There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

SR 3.3.7.4

SR 3.3.7.4 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and is performed every 18 months. Each Manual Actuation Function is tested through the BOP ESFAS.

The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

SR 3.3.7.5

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

BASES

SURVEILLANCE REQUIREMENTS (continued)

There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

SR 3.3.8.2

A COT is performed once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the EES actuation. ~~The setpoints shall be left consistent with the unit specific calibration procedure tolerance.~~ The Frequency of 92 days is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.8.3

SR 3.3.8.3 is the performance of an ACTUATION LOGIC TEST using the BOP ESFAS automatic tester. The actuation logic is tested every 31 days on a STAGGERED TEST BASIS. All possible logic combinations, with and without applicable permissive, are tested for each protection function. The Frequency is based on the known reliability of the relays and controls and the multichannel redundancy available, and has been shown to be acceptable through operating experience. The SR is modified by a Note stating that the continuity check may be excluded. This SR is applied to the balance of plant actuation logic and relays that do not have circuits installed to perform the continuity check.

SR 3.3.8.4

SR 3.3.8.4 is the performance of a TADOT. This test is a check of the manual actuation functions and is performed every 18 months. Each manual actuation function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (e.g., pump starts, valve cycles, etc.). The Frequency is based on operating experience and is consistent with the typical industry refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.5

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measure parameter within the necessary range and accuracy. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 100.11.
2. Calculation J-G-SA02.
3. USAR Section 7.3.3 and Table 7.3-5.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The Pressurizer pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS total flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the ~~safety analysis limit~~ DNBR ~~as specified in the COLR~~ criteria. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power

BASES

APPLICABLE SAFETY ANALYSIS (continued)

distribution limits are satisfied per LCO 3.1.4, "Rod Group Alignment Limits; " LCO 3.1.5, "Shutdown Bank Insertion Limits;" LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit and RCS average temperature limit specified in the COLR correspond to the analytical limits used in the safety analyses, with allowance for measurement uncertainty.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction ~~in the generic~~ margin. The ~~generic~~ margins completely offset any rod bow penalties. ~~This is the margin between the correlation DNBR limit and the safety analysis limit DNBR. These limits are specified in the COLR. The applicable values of rod bow penalties are referenced in the USAR.~~

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

In order to provide adequate DNB margin, a review of the past RCS flow performance at WCGS was performed and a value of 376,000 gpm was determined for the minimum measured flow (MMF). A MMF of 376,000 gpm, that is specified in the COLR, bounds the calculated uncertainty of 3.6 % RCS flow, which was calculated for the RCS Flow-Cold Leg Elbow Tap Indication as discussed in WCAP-17602-P (Reference 2).

This LCO specifies limits on the monitored process variables - pressurizer pressure, RCS average temperature, and RCS total flow rate - to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, usually based on the maximum analyzed steam generator tube plugging, is retained in the TS LCO. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

The RCS total flow rate limit ~~contains a measurement error of 2.5%~~ based on performing a precision heat balance and using the result to normalize the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner.

The effect of any fouling that might bias the flow rate measurement shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The LCO numerical values for pressure, temperature, and flow rate specified in the COLR have been adjusted for instrument error.

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS total flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The installed flow instrumentation provides indication as a percentage of total flow rate based on the precision calorimetric heat balance. Plant procedures specify the percentage of the total flow rate required to meet the RCS total flow rate limit. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every 18 months after each refueling allows the installed RCS flow instrumentation to be normalized and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate. When performing a precision heat balance, the instrumentation used for determining steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi Δp in the calorimetric calculations shall be calibrated within 7 days prior to performing the heat balance.

The Frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 7 days after $\geq$ 95% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 95% RTP to obtain the stated RCS flow accuracies and the test is only a confirmation of SR 3.4.1.4. The Surveillance shall be performed within 7 days after reaching 95% RTP.

REFERENCES

1. USAR, Chapter 15.

2. WCAP-17602-P, Revision 0, "Westinghouse Setpoint Calculations for the Wolf Creek Generating Station Control, Protection, and Indication Systems," [Date TBD].

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. ~~After several hours~~, the ECCS flow is shifted to the hot leg recirculation phase to ~~provide a~~ backflush~~, which would reduce the boiling in the top of~~ the core ~~and any resulting boron precipitation~~.

Between 6.5 - 7.5 hours after accident initiation

prior to reaching the boric acid solubility limit

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

The ECCS flow paths consist of piping, valves, heat exchangers,and pumps such that water from the RWST can be injected into the RCS

BASES

A third turbine trip analysis is performed to

APPLICABLE SAFETY ANALYSES (continued)

crediting reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that the maximum RCS pressure does not exceed 110% of the design pressure. ~~All cases analyzed~~ demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the Overtemperature ΔT or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the atmospheric or condenser steam dump valves. The USAR Section 15.4 safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO.

The USAR safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs.

In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an A00 by reducing the setpoint of the Power Range Neutron Flux-High reactor trip function. For example, if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrwal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value. When the Moderator Temperature Coefficient (MTC) is positive, the reactor power may increase above the initial value during an RCS heatup event (e.g., turbine trip). Thus, for any number of inoperable MSSVs it is necessary to reduce the trip setpoint if a positive MTC may exist at partial power conditions, unless it is demonstrated by analysis that

MSSVs
B 3.7.1

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative analysis of the loss of load/turbine trip event with inoperable MSSVs assumed.

BASES

ACTIONS (continued)

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

A.1

In the case of only a single inoperable MSSV on one or more steam generators when the Moderator Temperature Coefficient is not positive, a reactor power reduction alone is sufficient to limit primary side heat generation such that overpressurization of the secondary side is precluded for any RCS heatup event. Furthermore, for this case there is sufficient total steam flow capacity provided by the turbine and remaining insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Therefore, Required Action A.1 requires an appropriate reduction in reactor power within 4 hours.

~~A sensitivity study (Ref. 7) was performed to analyze the loss of load/turbine trip event initiated from power levels based on Table 3.7.1-1 and assuming both beginning of life and end of life reactivity feedback conditions. The results of all cases studied showed that the secondary system peak pressure was maintained below 110% of the secondary system design pressure limit.~~

B.1 and B.2

In the case of multiple inoperable MSSVs on one or more steam generators, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Futhermore, for a single inoperable MSSV on one or more steam generators when the Moderator Temperature Coefficient is positive the reactor power may increase as a result of an RCS heatup event such that flow capacity of the remaining OPERABLE MSSVs is insufficient. The 4 hour Completion Time for Required Action B.1 is consistent with A.1. An additional 32 hours is allowed in Required Action B.2 to reduce the setpoints. The completion time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction,

BASES

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative analysis of the loss of load/turbine trip event with inoperable MSSVs assumed.

ACTIONS

B.1 and B.2 (continued)

operating experience in resetting all channels of protective function and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

~~A sensitivity study (Ref. 7) was performed to analyze the loss of load/turbine trip event initiated from power levels based on Table 3.7.1-1 and assuming both beginning of life and end of life reactivity feedback conditions. The results of all cases studied showed that the secondary system peak pressure was maintained below 110% of the secondary system design pressure limit.~~

Required Action B.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the Reactor Protection System trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation," provides sufficient protection.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

C.1 and C.2

If the Required Actions are not completed within the associated Completion Time, or if one or more steam generators have ≥ 4 inoperable MSSVs, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code (Ref. 5), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 6). According to Reference 6, the following tests are required:

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. USAR, Section 10.3.2.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.

3. USAR, Section 15.2.

4. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.

5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

6. ANSI/ASME OM-1-1987.

7. ~~AN-94-017 Rev. 0, "RETRAN-02 MSSV Analysis for ITIP 2625," M. L. Howard, May 1994.~~

For the analysis of the inadvertent operation of the ECCS during power operation event in Reference 4, credit is taken for operator action to open one of the four SG ARVs to control the average of the cold leg temperatures to approximately 557°F.

BASES

APPLICABLE SAFETY ANALYSES (continued)

In the accident analysis presented in Reference 2, the ARVs are assumed to be used by the operator to cool down the unit to RHR entry conditions for accidents accompanied by a loss of offsite power. The main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event in Reference 3, the operator is required to perform a RCS cooldown using two intact steam generators to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. For SG overfill resulting from SGTR, RCS cooldown to RHR entry conditions using intact SG ARVs is necessary to terminate primary to secondary break flow. The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to RHR conditions for this event and also for other accidents. Thus, the SGTR is the limiting event for the ARVs. The number of ARVs required to be OPERABLE to satisfy the SGTR accident analysis requirements is four. If a single failure of one occurs and another is associated with the ruptured SG, two ARVs would remain OPERABLE for heat removal and RCS cooldown, as discussed in Reference 3.

The ARVs are equipped with block valves in the event an ARV spuriously fails open or fails to close during use.

The ARVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Four ARV lines are required to be OPERABLE. One ARV line is required from each of four steam generators to ensure that at least two ARV lines are available to conduct a RCS cooldown following an SGTR, in which one steam generator becomes unavailable due to a SGTR, accompanied by a single, active failure of a second ARV line on an unaffected steam generator. The block valves must be OPERABLE to isolate a failed open ARV line.

Failure to meet the LCO can result in the inability to achieve subcooling, consistent with the assumptions used in the steam generator tube rupture analysis, to facilitate equalizing pressures between the Reactor Coolant System and the ruptured steam generator. Failure to meet the LCO can also impact the recovery capability following a SG overfill scenario.

An ARV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.4.2

The function of the block valve is to isolate a failed open or leaking ARV. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. The Frequency is acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 10.3.

2. USAR, Chapter 15.

3. USAR, Section 15.6.3.

4. USAR, Section 15.5.1.

BASES

ACTIONS (continued)

I.1

Condition I corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. 8). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are in accordance with the recommendations of Regulatory Guide 1.9 (Ref. 3), Regulatory Guide 1.108 (Ref. 9), and Regulatory Guide 1.137 (Ref. 10), as addressed in the USAR.

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. This minimum steady state output voltage of 3950 V is 95% of the nominal 4160 V output voltage. This value, which is 210 V above the minimum utilization voltage specified in ANSI C84.1 (Ref. 11), allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level . This value provides for the OPERABILITY of required loads as shown by load flow calculations in support of NRC Branch Technical Position PSB-1. These calculations have demonstrated that no end use loads will be adversely affected from sustained operation above the degraded voltage ~~allowable value~~ as specified in SR 3.3.5.3. The 3950 V is above the calculated ~~allowable value~~. The specified maximum steady state output voltage of 4320 V ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is no more than the maximum rated operating voltages. The specified minimum and maximum frequencies of the DG are 59.4 Hz and 60.6 Hz.

Nominal Trip Setpoint (NTSP)

NTSP

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate

Proposed COLR Changes (for information only)

1.0 CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT (COLR) for Wolf Creek Generating Station Cycle 20 has been prepared in accordance with the requirements of Technical Specification 5.6.5.

The core operating limits that are included in the COLR affect the following Technical Specifications:

- 2.1.1 Reactor Core Safety Limits
- 3.1.1 Shutdown Margin (SDM)
- 3.1.3 Moderator Temperature Coefficient (MTC)
- 3.1.4 Rod Group Alignment Limits
- 3.1.5 Shutdown Bank Insertion Limits
- 3.1.6 Control Bank Insertion Limits
- 3.1.8 PHYSICS TESTS Exceptions - MODE 2
- 3.2.1 Heat Flux Hot Channel Factor $(F_Q(Z))$ (F_Q Methodology)
- 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor $(F_{\Delta H}^N)$
- 3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)
- 3.3.1 Reactor Trip System (RTS) Instrumentation
- 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation
- 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
- 3.9.1 Boron Concentration

~~The portions of the Technical Specification Bases affected by the report are listed below:~~

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

WOLF CREEK
NUCLEAR OPERATING CORPORATION

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the subsections below:

2.1 Reactor Core Safety Limits (SL 2.1.1)

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 in Figure 2.1.

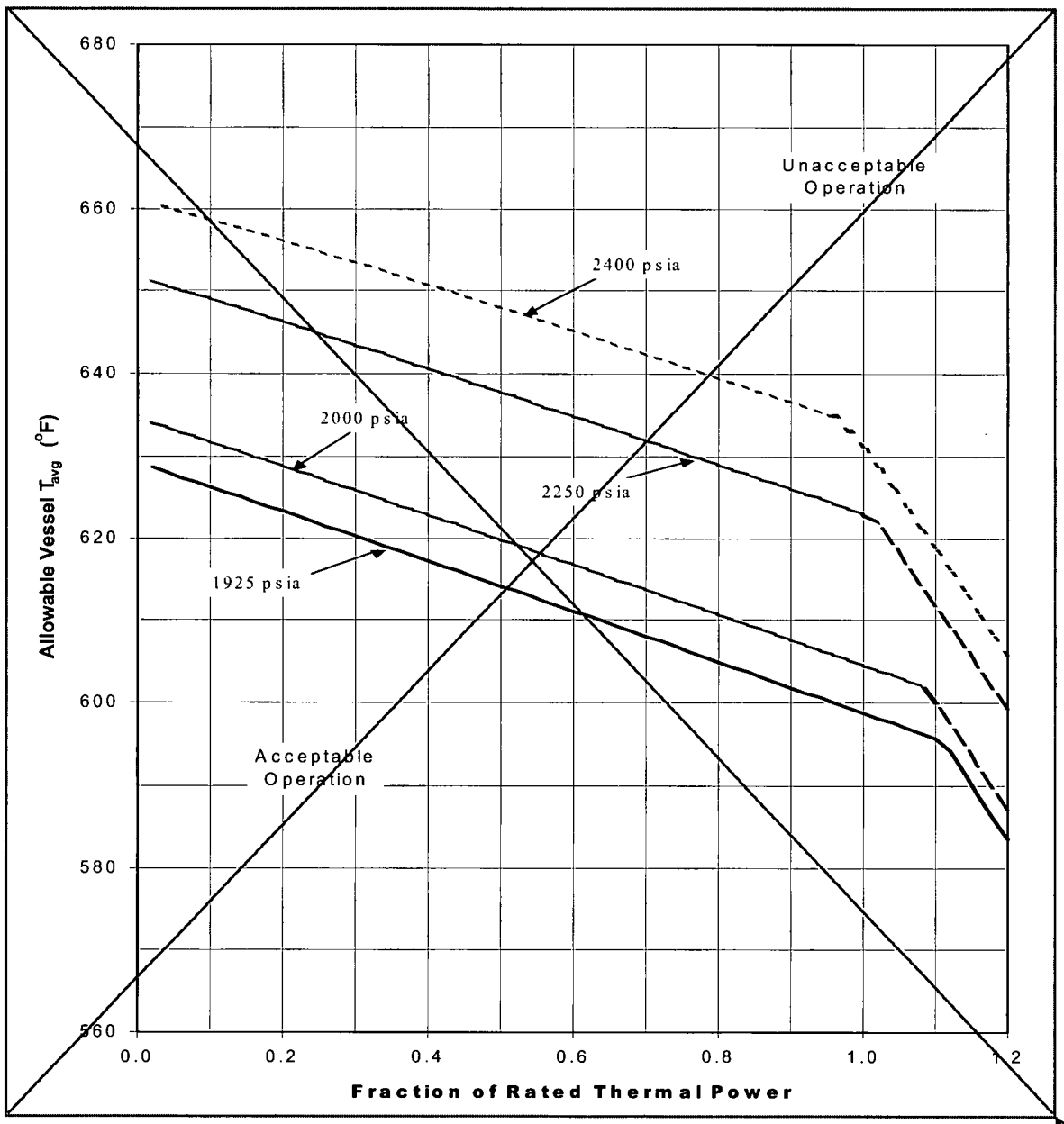
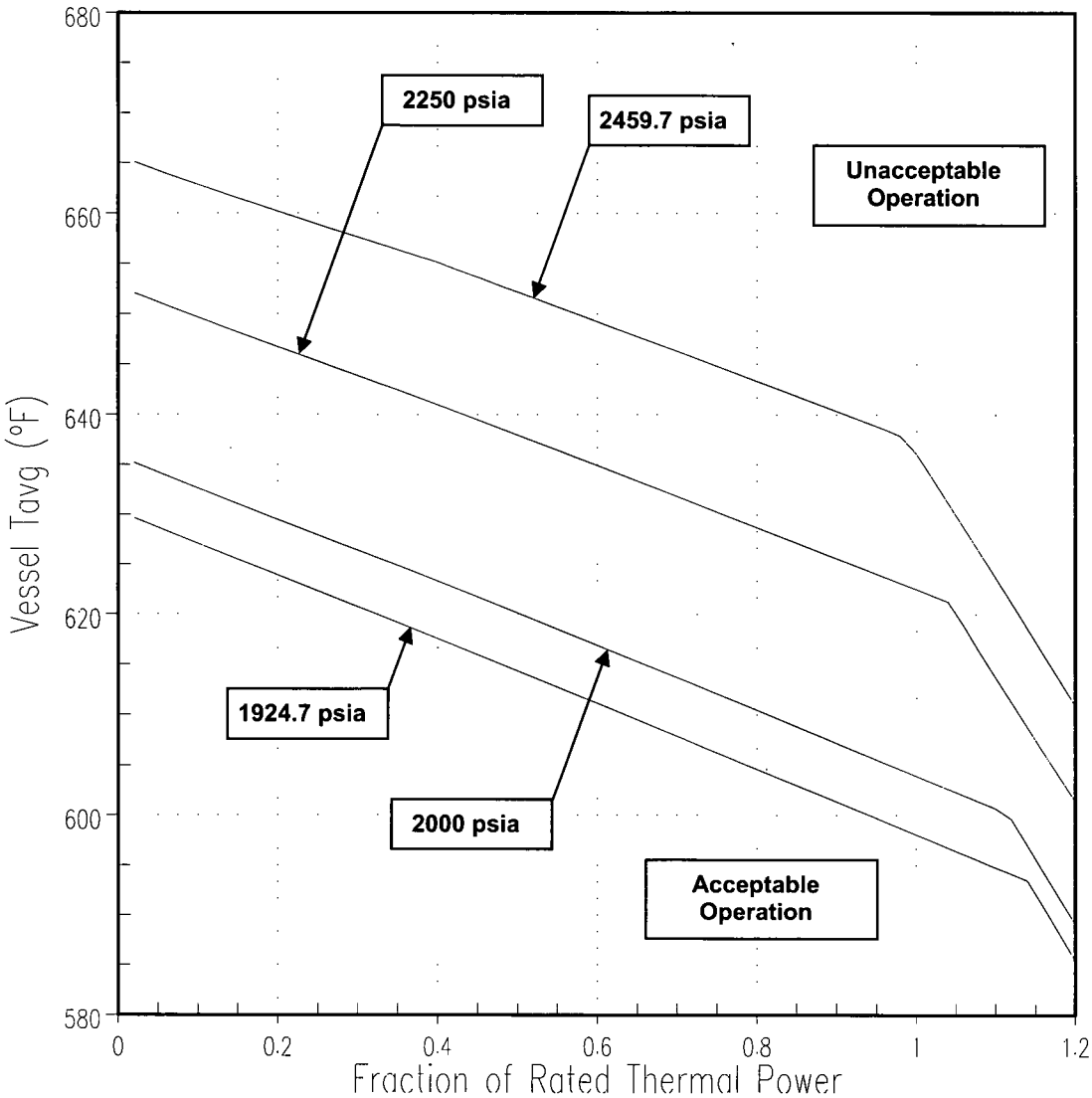


Figure 2.1
Reactor Core Safety Limits

Replace with new core limits figure attached.


680
660
640
620
600
580
Vessel Tavg (°F)
0
0.2
0.4
0.6
0.8
1
1.2
Fraction of Rated Thermal Power
2250 psia
2459.7 psia
Unacceptable Operation
1924.7 psia
2000 psia
Acceptable Operation

2.8 Reactor Trip System Overtemperature ΔT Setpoint Parameter Values (LCO 3.3.1)

Parameter	Value
Overtemperature ΔT reactor trip setpoint	$K_1 = 1.10$
Overtemperature ΔT reactor trip setpoint T_{avg} coefficient	$K_2 = 0.0137/°F$
Overtemperature ΔT reactor trip setpoint pressure coefficient	$K_3 =$ ~~0.000671/psig~~ 0.00095/psi
Nominal T_{avg} at RTP (T_{ref} from Rod Control)	$T' \leq 586.5°F$
Nominal RCS operating pressure	$P' \geq 2235$ psig
Measured RCS ΔT lead/lag constant	$\tau_1 = 6$ sec $\tau_2 = 3$ sec
Measured RCS ΔT lag constant	$\tau_3 = 2$ sec
Measured RCS average temperature lead/lag constant	$\tau_4 = 16$ sec $\tau_5 = 4$ sec
Measured RCS average temperature ~~lead/~~lag constant	$\tau_6 = 0$ sec

$f_1(\Delta I) = -0.0227$ / %RTP $\{23\% + (q_t - q_b)\}$ when $(q_t - q_b) < -23\%$ RTP

0% of RTP / %RTP when -23% RTP $\leq (q_t - q_b) \leq 5\%$ RTP

$0.0184 \{(q_t - q_b) - 5\%\}$ when $(q_t - q_b) > 5\%$ RTP

Where, q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

WOLF CREEK NUCLEAR OPERATING CORPORATION

Wolf Creek Generating Station
Cycle ~~20~~ 2X
Core Operating Limits Report
Revision 0

2.9 Reactor Trip System Overpower ΔT Setpoint Parameter Values (LCO 3.3.2)

Parameter	Value
Overpower ΔT reactor trip setpoint	$K_4 = 1.10$
Overpower ΔT reactor trip setpoint T_{avg} rate/lag coefficient	$K_5 = 0.02/°F$ for increasing T_{avg} $= 0/°F$ for decreasing T_{avg}
Overpower ΔT reactor trip setpoint T_{avg} heatup coefficient	$K_6 = 0.00128/°F$ for $T > T''$ $= 0/°F$ for $T \leq T''$
Nominal ~~Indicated~~ T_{avg} (T_{ref} from Rod Control) at RTP ~~(calibration temperature for ΔT instrumentation)~~	$T'' \leq 586.5°F$
Measured RCS ΔT lead/lag constant	$\tau_1 = 6$ sec $\tau_2 = 3$ sec
Measured RCS ΔT lag constant	$\tau_3 = 2$ sec
Measured RCS average temperature ~~lead/~~lag constant	$\tau_6 = 0$ sec
Measured RCS average temperature rate/lag constant	$\tau_7 = 10$ sec

$f_2(\Delta I) = 0\%$ RTP for all ΔI

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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)

592.5 °F * (Average of 4 channels)
592.3 °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.

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).

Delete Section 2.13

~~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~~

* This value is based on a full power nominal Tavg of 588.4 °F. If the full power Tavg for the operating cycle is defined at a different value within the analyzed full power Tavg window (570.7 °F to 588.4 °F), the DNB limit should be established by adding 4.1 °F to the cycle specific full power nominal Tavg value (when using the average of 4 channels) and by adding 3.9 °F to the cycle specific full power nominal Tavg value (when using the average of 3 channels).

B. Approved Analytical Methods for Determining Core Operating Limits

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." (ET 90-0140, ET 92-0103)~~

~~NRC Safety Evaluation Report dated October 29, 1992, for the "Core Thermal Hydraulic Analysis Methodology for the Wolf Creek Generating Station."~~

1 ~~2.~~ WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

NRC Safety Evaluation Report dated January 17, 1989, for the "Acceptance for Referencing of Licensing Topical Report WCAP-11397, Revised Thermal Design Procedure."

~~3. WCNOC Topical Report NSAG-006, "Transient Analysis Methodology for the Wolf Creek Generating Station" (ET-91-0026, ET 92-0142, WM 93-0010, WM 93-0028).~~

~~NRC Safety Evaluation Report dated September 30, 1993, for the "Transient Analysis Methodology for the Wolf Creek Generating Station."~~

~~EPRI Topical Report NP-7450(A), "RETRAN-3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," including NRC Safety Evaluation Report dated January 25, 2001, "Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," (TAC No. MA4311)." RETRAN-3D code is only utilized in the RETRAN-02 mode.~~

2 ~~4.~~ WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," February 1994.

NRC Safety Evaluation Report dated November 26, 1993, "Acceptance for Referencing of Revised Version of Licensing Topical Report WCAP-10216-P, Rev. 1, Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification" (TAC No. M88206).

3 ~~5.~~ ~~WCNOC Topical Report NSAG-007, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station" (ET 92-0032, ET 93-0017).~~

~~NRC Safety Evaluation Report dated March 10, 1993, for the "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" (NA 92-0073, NA 93-0013, NA 93-0054).~~

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

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."

4 ~~7~~. WCAP-10266-P-A, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987.

NRC letter dated November 13, 1986, "Acceptance for Referencing of Licensing Topical Report WCAP-10266 "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code.""

WCAP-10266-P-A, Addendum 1, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code Addendum 1: Power Shape Sensitivity Studies," December 1987.

NRC letter dated September 15, 1987, "Acceptance for Referencing of Addendum 1 to WCAP-10266, BASH Power Shape Sensitivity Studies."

WCAP-10266-P-A, Addendum 2, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code Addendum 2: BASH Methodology Improvements and Reliability Enhancements," May 1988

5 NRC letter dated January 20, 1988, "Acceptance for Referencing Topical Report Addendum 2 to WCAP-10266, Revision 2, "BASH Methodology Improvements and Reliability Enhancements."

~~8~~. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988.

6 NRC Safety Evaluation Report dated May 17, 1988, "Acceptance for Referencing of Westinghouse Topical Report WCAP-11596 - Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores."

~~9~~. WCAP 10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1988.

7 NRC letter dated June 23, 1986, "Acceptance for Referencing of Topical Report WCAP 10965-P and WCAP 10966-NP."

~~10~~. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995.

NRC Safety Evaluation Reports dated July 1, 1991, "Acceptance for Referencing of Topical Report WCAP-12610, 'VANTAGE+ Fuel Assembly Reference Core Report' (TAC NO. 77258)."

8 NRC Safety Evaluation Report dated September 15, 1994, "Acceptance for Referencing of Topical Report WCAP-12610, Appendix B, Addendum 1, 'Extended Burnup Fuel Design Methodology and ZIRLO Fuel Performance Models' (TAC NO. M86416)."

~~11~~. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Function." September 1986.

NRC Safety Evaluation Report dated April 17, 1986, "Acceptance for Referencing of Licensing Topical Report WCAP-8745(P)/8746(NP), 'Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions.'"