Part 4 Technical Specifications and Bases

Introduction

The U.S. EPR generic Technical Specifications and Bases, provided in Chapter 16 of the U.S. EPR FSAR are used to develop the {CCNPP Unit 3} Technical Specifications and Bases. Departures from the U.S. EPR generic Technical Specifications and Bases include replacing preliminary information provided in the brackets of the generic Technical Specifications and Bases with {CCNPP Unit 3} specific values and information, addressing and removing Reviewer's Notes, correcting typographical and editorial errors, and use of a Setpoint Control Program. These departures are addressed and justified in Section A of this part of the COL Application. Section B provides a complete copy of the {CCNPP Unit 3} Technical Specifications and Bases.

Section A – Departures

1. Typographical Error Correction – TS 1.1

    GTS:

    GTS 1.1, "Definitions," DOSE EQUIVALENT I-131, in referring to EPA Federal Guidance Report No .11, states "imiting Value of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

    {CCNPP Unit 3} TS:

    {CCNPP Unit 3} TS 1.1, "Definitions," DOSE EQUIVALENT I-131, in referring to EPA Federal Guidance Report No .11, is revised to state "Limiting Value of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

    Justification:

    The change corrects a typographical error in the GTS.

2. Setpoint Control Program

    Generic Technical Specifications (GTS) and Bases:

    GTS Table 3.3.1-2 includes a bracketed Reviewer's Note that states "The values specified in brackets in the Limiting Trip Setpoint column are included for reviewer information only. A plant specific setpoint study will be conducted. The values of the Limiting Trip Setpoint will then be replaced after the completion of the study."

    {CCNPP Unit 3} Technical Specifications (TS) and Bases:

    In order to address this bracketed Reviewer's Note, a Setpoint Control Program is adopted in the {CCNPP Unit 3} TS. TS 5.5.18, Setpoint Control Program

(SCP), is added to the TS. The TS requirements for the Setpoint Control Program establishes that Limiting Trip Setpoints (LTSPs), Nominal Trip Setpoints (NTSPs), Allowable Values (AVs), and As-Found Tolerance and As-Left Tolerance Bands for each of the required Technical Specification Instrument Functions in TS 3.3.1, “Protection Systems (PS),” shall be documented in the SCP. The TS requirements for the SCP also establish that the methods used to determine the Limiting Trip Setpoints (LTSPs), Nominal Trip Setpoints (NTSPs), Allowable Values (AVs), and As-Found Tolerance and As-Left Tolerance Bands for the required instrument functions shall be those included in NRC approved setpoint methodology documents. These NRC approved setpoint methodology documents are listed in TS 5.5.18. The TS requirements for the SCP also include the Technical Specification Task Force (TSTF)-493, “Clarify Application of Setpoint Methodology for LSSS Functions,” guidance to provide assurance that the required instruments will always actuate safety functions at the point assumed in the applicable safety analyses. Finally, the TS for the SCP require the SCP to be provided, including any revisions or supplements, to the NRC on a periodic basis.

To implement the SCP, the following additional changes to GTS and Bases 3.3.1 are made in the {CCNPP Unit 3} TS and Bases 3.3.1:

a. LTSPs specified in GTS Table 3.3.1-2 are replaced with a Setting Basis (i.e., the Analytical Limit or Design Limit, as defined in the Bases for each of the applicable instrument functions) in {CCNPP Unit 3} Table 3.3.1-2. The LTSPs are relocated to the SCP (TS 5.5.18).
b. The reference to “Limiting Trip Setpoint” in GTS 3.3.1, Condition C, is replaced with a reference to “Setpoint Control Program requirements” in {CCNPP Unit 3} TS 3.3.1, Condition C.
c. GTS SRs 3.3.1.4 and 3.3.1.6, which state “Perform CALIBRATION,” are revised in {CCNPP Unit 3} SRs 3.3.1.4 and 3.3.1.6 to “Perform CALIBRATION consistent with Specification 5.5.18, Setpoint Control Program (SCP).”
d. GTS Table 3.3.1-2 footnotes (b) and (c), which implement TSTF-493 guidance, are moved to {CCNPP Unit 3} TS 5.5.18.c.1 and 5.5.18.c.2 and the subsequent footnotes in {CCNPP Unit 3} Table 3.3.1-2 are relabeled.
e. The GTS Table 3.3.1-2 bracketed Reviewer’s Note is deleted from {CCNPP Unit 3} Table 3.3.1-2.
f. Corresponding changes to the GTS Bases 3.3.1, including deletion of the associated Reviewer’s Notes, are made in the {CCNPP Unit 3} Bases 3.3.1.

Justification:

Table 3.3.1-2 contains a Reviewer’s Note which requires a plant specific setpoint study to be conducted and that the values of the Limiting Trip Setpoint be replaced after the completion of the study. However, the plant specific setpoint study can not be completed until after selection of instrumentation. Nevertheless, instrumentation selection may not occur until after the approval of the COL application is granted. As an alternative approach, it is proposed that the Limiting Trip Setpoints be relocated to the Setpoint Control Program and that the Setting Basis (Analytical Limits or Design Limits, as applicable) for the required instrument functions be specified in the TS. The Setpoint Control

Program is a TS required program and is consistent with the approach used for the TS required Core Operating Limits Report and the Pressure and Temperature Limits Report. In the case of the Core Operating Limits, the NRC approved relocation of cycle-specific parameter limits from the TS to the Core Operating Limits Report. The basis for acceptability of this approach was that the methodology for determining cycle-specific parameter limits is documented in NRC approved topical reports or in an NRC approved plant-specific submittal. As a consequence the NRC review of proposed changes to the TS for these cycle-specific parameter limits was primarily limited to confirmation that the updated limits were calculated using an NRC approved methodology and consistent with applicable limits of the safety analysis. The approach documented in the TS for the Core Operating Limits Report also allows the NRC to trend the parameter limit changes, if desired. The Core Operating Limits Report approach is documented in NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits for Technical Specifications," dated October 3, 1988, and is reflected in the current Improved Standard Technical Specifications (NUREG-1430 through NURG-1434). For the Setpoint Control Program, the TS require that that the Limiting Trip Setpoints be developed using NRC approved setpoint methodology. In addition, by specifying the Analytical Limits and Design Limits in the TS, assurance is provided that the Limiting Trip Setpoints are developed and maintained such that required instruments will always actuate safety functions at the point assumed in the applicable safety analyses. The approach documented in the TS for the Setpoint Control Program also allows the NRC to trend the parameter limit changes, if desired, since the TS requires the Setpoint Control Program to be submitted to the NRC prior to initial fuel load and periodically thereafter.

3. Error Correction – TS 3.1.1 Limiting Trip Setpoint Inequality Signs

<u>GTS:</u>

GTS 3.3.1, "Protection System (PS)," Table 3.3.1-2, includes Functions with Limiting Trip Setpoint values which are either missing inequality signs or have inequality signs specified incorrectly. These PS Functions are as follows.

Function A.3,	High Neutron Flux Rate of Change (Power Change)
Function A.5,	Low Saturation Margin
Function A.14,	Steam Generator (SG) Pressure Drop
Function A.17,	Low SG Level
Function A.18,	High SG Level
Function A.19,	High Containment Pressure
Function B.2.b,	Main Feedwater Full Load Closure on High SG Level (Affected SGs)
Function B.2.c,	Startup and Shutdown Feedwater Isolation on SG Pressure Drop (All SGs)
Function B.2.e,	Startup and Shutdown Feedwater Isolation on High SG Level for Period of Time (Affected SGs)
Function B.8.a,	Main Steam Isolation Valve (MSIV) Closure on SG Pressure Drop (All SGs)
Function B.9.a,	Containment Isolation (Stage 1) on High Containment Pressure

Function B.9.c, Containment Isolation (Stage 2) on High-High Containment Pressure
Function B.9.d, Containment Isolation (Stage 1) on High Containment Radiation

{CCNPP Unit 3} TS:

The Setting Basis values for the following Functions in {CCNPP Unit 3} TS 3.3.1, "Protection System (PS)," Table 3.3.1-2, are revised to include the missing inequality signs and to correct the inequality signs, as required. Corresponding changes are made to the Bases, as required.

Function A.3, High Neutron Flux Rate of Change (Power Change)
Function A.5, Low Saturation Margin
Function A.14, Steam Generator (SG) Pressure Drop
Function A.17, Low SG Level
Function A.18, High SG Level
Function A.19, High Containment Pressure
Function B.2.b, Main Feedwater Full Load Closure on High SG Level (Affected SGs)
Function B.2.c, Startup and Shutdown Feedwater Isolation on SG Pressure Drop (All SGs)
Function B.2.e, Startup and Shutdown Feedwater Isolation on High SG Level for Period of Time (Affected SGs)
Function B.8.a, Main Steam Isolation Valve (MSIV) Closure on SG Pressure Drop (All SGs)
Function B.9.a, Containment Isolation (Stage 1) on High Containment Pressure
Function B.9.c, Containment Isolation (Stage 2) on High-High Containment Pressure
Function B.9.d, Containment Isolation (Stage 1) on High Containment Radiation

Justification:

The change corrects errors in the GTS to be consistent with the U.S. EPR design and analyses. These errors will be corrected in the GTS in a future revision.

4. Error Correction – TS 3.1.1 Time Delays

GTS:

GTS 3.3.1, "Protection System (PS)," Table 3.3.1-2 includes Limiting Trip Setpoint values with time delays for Function A.18, High SG Level, and Function B.2.b, Main Feedwater Full Load Closure on High SG Level (Affected SGs).

{CCNPP Unit 3} TS:

The time delays are removed from the {CCNPP Unit 3} TS 3.3.1, "Protection System (PS)," Table 3.3.1-2 Setting Basis values for Function A.18, High SG Level, and Function B.2.b, Main Feedwater Full Load Closure on High SG Level (Affected SGs).

Justification:

The change corrects errors in the GTS to be consistent with the U.S. EPR design and analyses. These errors will be corrected in the GTS in a future revision.

5. Error Correction – Location of Limiting Trip Setpoint/Setting Basis value in Core Operating Limits Report (COLR)

GTS:

GTS 3.3.1, "Protection System (PS)," Table 3.3.1-2 includes a Limiting Trip Setpoint value for Function B.11.b, CVCS Charging Line Isolation on Anti-Dilution Mitigation (ADM) at Shutdown Condition (RCP not operating).

{CCNPP Unit 3} TS:

The Setting Basis for {CCNPP Unit 3} TS 3.3.1, "Protection System (PS)," Table 3.3.1-2 Function B.11.b, CVCS Charging Line Isolation on Anti-Dilution Mitigation (ADM) at Shutdown Condition (RCP not operating), is revised to indicate that the value is "As specified in the COLR."

Justification:

The change corrects an error in the GTS. The values associated with the Limiting Trip Setpoint and Setting Basis are cycle-specific parameter values. As such, consistent with the Limiting Trip Setpoint specified in GTS 3.3.1, Table 3.3.1-2 for Function B.11.c, CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions, it is appropriate for the Limiting Trip Setpoint and Setting Basis for Function B.11.b to also be specified in the COLR. This error will be corrected in the GTS in a future revision.

6. Typographical Error Correction – TS 3.5.3

GTS:

GTS 3.5.3, "ECCS – Shutdown," Required Action B.1 states "Be in Mode 5."

{CCNPP Unit 3} TS:

{CCNPP Unit 3} TS 3.5.3, "ECCS – Shutdown," Required Action B.1 is revised to state "Be in MODE 5."

Justification:

The word "Mode" is a defined term in TS Section 1.1, "Definitions," and therefore is required to appear in all capitalized type in the TS and Bases in accordance with the Note at the beginning of TS Section 1.1.

7. {Toxic Gas/Hazardous Chemical Protection

GTS and Bases:

GTS 3.7.10, “Control Room Emergency Filtration (CREF),” Required Action B.1, Required Action D.1, and associated Reviewer’s Note, Bases 3.7.10 (Background, Applicable Safety Analyses, LCO, Actions sections), Bases 3.7.12, “Safeguard Building Controlled Area Ventilation System (SBVS),” Action B.1, and GTS 5.5.17, “Control Room Envelope Habitability Program,” contain conceptual design information on toxic gas and hazardous chemicals.

{CCNPP Unit 3} TS:

The information in CCNPP Unit 3 TS 3.7.10, “Control Room Emergency Filtration (CREF),” Required Action B.1, Required Action D.1, and associated Reviewer’s Note, Bases 3.7.10 (Background, Applicable Safety Analyses, LCO, Actions and Surveillance Requirements section, sections), Bases 3.7.12, “Safeguard Building Controlled Area Ventilation System (SBVS),” Action B.1, and TS 5.5.17, “Control Room Envelope Habitability Program,” pertaining to toxic gas and hazardous chemicals is removed.

Justification:

The conceptual design information and Reviewer’s Note associated with toxic gas and hazardous chemical protection are not applicable in CCNPP Unit 3. Toxic gas and hazardous chemical protection for the CREF is not required for CCNPP Unit 3 based on the site-specific evaluation provided in CCNPP Unit 3 FSAR Sections 2.2.3 and 6.4.4.}

8. Spent Fuel Storage Pool Boron Concentration

GTS:

GTS and Bases 3.7.15, “Spent Fuel Storage Pool Boron Concentration,” includes a Reviewer’s Note that states “The design of the spent fuel storage racks is to be provided by the COLA applicant. The required boron concentration will be provided as part of the spent fuel rack design.”

{CCNPP Unit 3} TS:

{CCNPP Unit 3} TS and Bases 3.7.15, “Spent Fuel Storage Pool Boron Concentration,” are revised to remove the Reviewer’s Note and include the {CCNPP Unit 3} specific spent fuel storage pool boron concentration in the TS and Bases. The associated Bases (all sections) are also revised to include a discussion of the bases for the specific spent fuel storage pool boron concentration requirements.

Justification:

The {CCNPP Unit 3} specific TS and Bases 3.7.15 are revised to reflect the results of the analyses of the {CCNPP Unit 3} specific spent fuel rack design described in the Holtec Topical Report for the design and analyses of the U.S. EPR spent fuel storage racks.

9. Spent Fuel Storage

GTS:

GTS and Bases 3.7.16, "Spent Fuel Storage," includes a Reviewer's Note that states "The design of the spent fuel storage racks is to be provided by the COLA applicant. The required spent fuel storage configuration will be provided as part of the spent fuel rack design."

{CCNPP Unit 3} TS:

{CCNPP Unit 3} TS and Bases 3.7.16, "Spent Fuel Storage," are revised to remove the Reviewer's Note and include the {CCNPP Unit 3} specific spent fuel storage requirements in the TS and Bases, including the addition of Figure 3.7.16-1, "Fuel Assembly Burnup Requirements for Region 2." The associated Bases (all sections) are also revised to include a discussion of the bases for the specific spent fuel storage requirements.

Justification:

The {CCNPP Unit 3} specific TS and Bases 3.7.16 are revised to reflect the results of the design and analyses of the {CCNPP Unit 3} specific spent fuel racks described in the Holtec Topical Report for the design and analyses of the U.S. EPR spent fuel storage racks.

10. Site Location

GTS:

GTS 4.1, "Site Location," contains a bracketed requirement for the COL application to provide site specific information for Section 4.1, "Site Location."

{CCNPP Unit 3} TS:

The bracketed information in {CCNPP Unit 3} TS 4.1, Site Location," is removed and replace the information with {CCNPP Unit 3} specific information regarding the site location.

Justification:

The site location information provided is consistent with the {CCNPP Unit 3} FSAR description of site location.

11. Fuel Storage Rack Uncertainties

GTS:

GTS 4.3.1, “Criticality,” includes a Reviewer’s Note that states “Storage rack uncertainties are discussed in the FSAR of COLA Section 9.1.” In addition, GTS 4.3.1.1.b, 4.3.1.2.b, and 4.3.1.2.c include brackets around the reference (i.e., FSAR) for the location of the description of the spent and new fuel storage rack uncertainties.

{CCNPP Unit 3} TS:

The Reviewers Note in {CCNPP Unit 3} TS 4.3.1, Criticality,” is removed and the brackets are removed from around the GTS reference for the location of the description of the spent and new fuel storage rack uncertainties (i.e., FSAR).

Justification:

{CCNPP Unit 3} FSAR Section 9.1 will include, by incorporation of reference, the description of spent and new fuel storage rack uncertainties.

12. Spent Fuel Storage Dimensions and Restrictions

GTS:

GTS 4.3.1, “Criticality,” Section 4.3.1.1 contains bracketed requirements for the COL application to provide the spent fuel rack center to center distance between fuel assemblies.

{CCNPP Unit 3} TS:

{CCNPP Unit 3} TS 4.3.1.1.c is revised to include the spent fuel rack center to center distances between fuel assemblies stored in Region 1 and for fuel assemblies stored in Region 2 of the {CCNPP Unit 3} spent fuel racks. In addition, restrictions on the storage of fuel assemblies are established in TS 4.3.1.1.d and 4.3.1.1.e. Figure 4.3-1, “Discrete Two Region Spent Fuel Pool Rack Layout” is added to identify the storage cells associated with Region 1 and the storage cells associated with Region 2.

Justification:

{CCNPP Unit 3} specific TS 4.3.1.1 requirements are revised to reflect the results of the design and analyses of the {CCNPP Unit 3} specific spent fuel racks described in the Holtec Topical Report for the design and analyses of the U.S. EPR spent fuel storage racks.

13. New Fuel Storage Dimensions

GTS:

GTS 4.3.1, “Criticality,” Section 4.3.1.2 contains bracketed requirements for the COL application to provide the new fuel rack center to center distance between fuel assemblies.

{CCNPP Unit 3} TS:

{CCNPP Unit 3} TS 4.3.1.2.d is revised to include the new fuel rack center to center distances between fuel assemblies stored in the {CCNPP Unit 3} new fuel racks.

Justification:

{CCNPP Unit 3} specific TS 4.3.1.2 requirements are revised to reflect the results of the design and analyses of the {CCNPP Unit 3} specific new fuel racks described in the Holtec Topical Report for the design and analyses of the U.S. EPR new fuel storage racks.

14. Spent Fuel Storage Capacity

GTS:

GTS 4.3.3, “Capacity,” contains bracketed requirements for the COL application to provide the capacity for spent fuel storage in the spent fuel storage pool.

{CCNPP Unit 3} TS:

{CCNPP Unit 3} TS 4.3.3 is revised to include capacity for spent fuel storage in the {CCNPP Unit 3} spent fuel storage pool.

Justification:

{CCNPP Unit 3} specific TS 4.3.3 requirements are revised to reflect the spent fuel storage capacity resulting from the design and analyses of the {CCNPP Unit 3} specific spent fuel racks described in the Holtec Topical Report for the design and analyses of the U.S. EPR spent fuel storage racks.

15. Generic Organizational Titles

GTS:

GTS 5.1, “Responsibility,” includes two Reviewer’s Notes related titles for members of the unit staff.

{CCNPP Unit 3} TS:

{CCNPP Unit 3} TS 5.1 is revised to remove the Reviewer’s Notes and replace them with a note requiring that the organizational positions listed in the Administrative Controls section have corresponding plant-specific titles specified in the Final Safety Analysis Report (FSAR).

Justification:

The use of generic titles in the TS, and the inclusion of plant-specific, corresponding titles in the FSAR, is consistent with Improved Standard Technical Specifications, Revision 3.1 of NUREG-1430 through NUREG-1434.

16. Non-licensed Operators for Two Units

GTS:

GTS 5.2.2, “Unit Staff,” contains a Reviewer’s Note specifying the number of non-licensed operators required for two units when both units are shutdown or defueled.

{CCNPP Unit 3} TS:

{CCNPP Unit 3} TS 5.2.2, “Unit Staff,” is revised to remove the Reviewer’s Note.

Justification:

The {CCNPP Unit 3} U.S. EPR is a single unit facility.

17. Minimum Qualifications of Unit Staff

GTS:

GTS 5.3, “Unit Staff Qualifications,” contains a Reviewer’s Note on the specification of the minimum qualifications of the unit staff.

{CCNPP Unit 3} TS:

{CCNPP Unit 3} TS 5.3, “Unit Staff Qualifications,” is revised to remove the Reviewer’s Note.

Justification:

The unit staff qualifications standards are provided consistent with the {CCNPP Unit 3} FSAR, including FSAR Section 13.2, for the stated exception regarding cold license operator candidates.

18. Temporary Outdoor Liquid Radwaste Storage Tanks

GTS:

GTS 5.11, “Gaseous Waste Processing System Radioactivity Monitoring Program,” contains a Reviewer’s Note for applicants incorporating outdoor liquid radioactive waste storage tanks in their design.

{CCNPP Unit 3} TS:

{CCNPP Unit 3} TS 5.11, “Gaseous Waste Processing System Radioactivity Monitoring Program,” is revised to remove the Reviewer’s Note.

Justification:

The {CCNPP Unit 3} specific design does not include outdoor liquid radioactive waste storage tanks.

19. Containment Bypass Leakage Paths

GTS:

GTS 5.15, “Containment Leakage Rate Testing Program,” contains a Reviewer’s Note indicating that, as discussed in FSAR Section 6.2.6, the U.S. EPR has no penetrations that are classified as bypass leakage paths.

{CCNPP Unit 3} TS:

{CCNPP Unit 3} TS 5.15, “Containment Leakage Rate Testing Program,” is revised to remove the Reviewer’s Note.

Justification:

The {CCNPP Unit 3} specific design has no penetrations that are classified as bypass leakage paths. This design information is reflected in FSAR Section 6.2.6 and does not need to be repeated in the TS.

20. {Multi-Unit Site Reporting Options

GTS:

GTS 5.6.1, “Annual Radiological Environmental Operating Report,” and GTS 5.6.2, “Radioactive Effluent Release Report” contain Reviewer’s Notes to allow a single report submittal for all units at a multi-unit site.

CCNPP Unit 3 TS:

CCNPP Unit 3 TS 5.6.1, “Annual Radiological Environmental Operating Report,” and GTS 5.6.2, “Radioactive Effluent Release Report” are revised to remove the Reviewer’s Notes.

Justification:

The allowance for submittal of a single report for all units at the CCNPP site does not apply to CCNPP Unit 3 since CCNPP Unit 3 is operated by a separate company from the company that operates CCNPP Units 1 and 2}.

21. Editorial Error Correction – Bases 3.3.1

GTS Bases:

GTS Bases 3.3.1, "Protection System (PS)," in the Applicable Safety Analyses, LCO, and Applicability section includes a discussion of Function A.20, Manual Reactor Trip. However, GTS 3.3.1, "Protection System (PS)," does not include a Function A.20 and does not include requirements for a Manual Reactor Trip Function.

{CCNPP Unit 3} TS Bases:

CCNPP Unit 3 Bases 3.3.1, "Protection System (PS)," in the Applicable Safety Analyses, LCO, and Applicability section is revised to eliminate the discussion of Function A.20, Manual Reactor Trip.

Justification:

The removal of the Bases discussion of a trip function (i.e., Manual Reactor Trip) that does not appear in the Technical Specifications is considered a correction of an editorial error. This change does not result in a change to the GTS and does not change the intent of the GTS.

22. Editorial Error Correction – Bases 3.3.1

GTS Bases:

GTS Bases 3.3.1, "Protection System (PS)," in the Applicable Safety Analyses, LCO, and Applicability section includes a discussion of the Limiting Trip Setpoint (LTSP) for Function B.5, Partial Cooldown on SIS Actuation. However, GTS 3.3.1, "Protection System (PS)," does not include an LTSP for Function B.5. GTS Table 3.3.1-2 (page 3 of 6) for Function B.5 states that the LTSP for this function is "NA."

CCNPP Unit 3 TS Bases:

CCNPP Unit 3 Bases 3.3.1, "Protection System (PS)," in the Applicable Safety Analyses, LCO, and Applicability section is revised to eliminate the discussion of the LTSP for Function B.5, Partial Cooldown on SIS Actuation.

Justification:

The removal of the Bases discussion of the LTSP for a Function that does not include an LTSP in the Technical Specifications is considered a correction of an editorial error. This change does not result in a change to the GTS and does not change the intent of the GTS.

23. Typographical Error Corrections – Bases 3.3.1

GTS:

GTS Bases 3.3.1, "Protection System (PS)," in the Applicable Safety Analyses, LCO, and Applicability section states the following.

- For Function A.2, High Linear Power Density, in the fourth paragraph, the last sentence states, in part, "…to unacceptable fuel centerline melt for any AOOs lead to an uncontrolled increase of the linear power density."
- For Function B.10.a, Emergency Diesel Generator (EDG) Start on Degraded Grid Voltage, the fourth paragraph states, in part, "…power is provide to ESF Functions…"
- For Function B.10.b, EDG Start on LOOP, the fourth paragraph states, in part, "…power is provide to ESF Functions…"

GTS Bases 3.3.1, "Protection System (PS)," in the Surveillance Requirements section, in the first sentence of the first paragraph of the discussion of SR 3.3.1.1, refers to "SR 3.3.1.2."

{CCNPP Unit 3} TS:

{CCNPP Unit 3} Bases 3.3.1, "Protection System (PS)," in the Applicable Safety Analyses, LCO, and Applicability section is revised as follows.

- For Function A.2, High Linear Power Density, in the fourth paragraph, revise the last sentence to read, in part, "…to unacceptable fuel centerline melt for any AOOs that lead to an uncontrolled increase of the linear power density."
- For Function B.10.a, Emergency Diesel Generator (EDG) Start on Degraded Grid Voltage, revise the fourth paragraph to read, in part, "…power is provided to ESF Functions…"
- For Function B.10.b, EDG Start on LOOP, revise the fourth paragraph to read, in part, "…power is provided to ESF Functions…"

{CCNPP Unit 3} Bases 3.3.1, "Protection System (PS)," in the Surveillance Requirements section, in the first sentence of the first paragraph of the discussion of SR 3.3.1.1, is revised to refer to "SR 3.3.1.1."

Justification:

The change corrects typographical errors in the GTS.

24. Application of Topical Reports for Surveillance Frequency Extensions

GTS:

GTS Bases 3.3.1, “Protection System (PS),” in the Surveillance Requirements section, includes a Reviewer’s Note that states “In order for a plant to take credit for topical reports as the basis for justifying Frequencies, topical reports must be supported by an NRC staff SER that establishes the acceptability of each topical report for that unit.”

{CCNPP Unit 3} TS:

{CCNPP Unit 3} Bases 3.3.1, “Protection System (PS),” in the Surveillance Requirements section, is revised to remove the Reviewer’s Note.

Justification:

The specified Frequencies in the {CCNPP Unit 3} TS 3.3.1 are based on the Frequencies specified in GTS 3.3.1. {CCNPP Unit 3} TS 3.3.1 does not credit topical reports as the basis for justifying Surveillance Frequencies.

25. Containment Leakage Acceptance Criteria

GTS Bases:

GTS Bases 3.6.1, “Containment,” contains a Reviewer’s Note, in the Bases for SR 3.6.1.1 indicating that Regulatory Guide 1.163 and NEI 94-01 contain acceptance criteria for containment leakage which may be reflected in the Bases.

{CCNPP Unit 3} TS Bases:

{CCNPP Unit 3} TS Bases 3.6.1, “Containment,” is revised to remove the Reviewer’s Note.

Justification:

The {CCNPP Unit 3} Containment Leakage Rate Testing Program is conducted as required by TS 5.15, “Containment Leakage Rate Testing Program,” and U.S. EPR FSAR 6.2.6, “Containment Leakage Testing.” U.S. EPR FSAR 6.2.6 was developed to be consistent with Regulatory Guide 1.163 and NEI 94-01. Therefore, the information reflected in the Reviewer’s Note does not need to be included in the Bases.

26. Seismic Category 1 Essential Service Water System Makeup

GTS Bases:

GTS Bases 3.7.8, “Essential Service Water (ESW) System,” contains a bracketed requirement in the Background section for the COL application to provide site specific information for the seismic Category 1 to the ESW System.

{CCNPP Unit 3} TS Bases:

{CCNPP Unit 3} Bases 3.7.8, “Essential Service Water (ESW) System,” is revised, in the Background section, to remove the bracketed requirement and replace the information with {CCNPP Unit 3} specific information regarding the seismic Category 1 ESW System makeup.

Justification:

The site-specific information provided is consistent with the {CCNPP Unit 3} FSAR Section 9.2 description of seismic Category 1 ESW System makeup.

27. Definition of OPERABLE Essential Service Water System Makeup Source

GTS Bases:

GTS Bases 3.7.8, “Essential Service Water (ESW) System,” contains a bracketed requirement in the LCO section for the COL application to provide site specific information for the definition of an OPERABLE ESW System makeup source.

{CCNPP Unit 3} TS Bases:

{CCNPP Unit 3} Bases 3.7.8, “Essential Service Water (ESW) System,” is revised, in the LCO section, to remove the bracketed requirement and replace the information with {CCNPP Unit 3} specific information regarding the definition of an OPERABLE ESW System makeup source.

Justification:

The site specific information provided is consistent with the {CCNPP Unit 3} FSAR Section 9.2 description of seismic Category 1 ESW System makeup source.

28. Typographical Error Correction – Bases 3.7.8

GTS Bases:

GTS Bases 3.7.8, “Essential Service Water (ESW) System,” in the Bases for SR 3.7.8.2, includes brackets around the ESW cooling tower basin temperature limit of 90°F.

{CCNPP Unit 3} TS Bases:

{CCNPP Unit 3} TS 3.7.8, “Essential Service Water (ESW) System,” is revised to remove the brackets from around the ESW cooling tower basin temperature limit of 90°F

Justification:

Throughout GTS and Bases 3.7.8, the ESW cooling tower basin limit of 90°F is not bracketed except in this one location in the Bases of SR 3.7.8.2. Therefore, the inclusion of brackets around this one occurrence of the 90°F ESW cooling tower basin limit is considered to be a typographical error. In addition, the analyses performed for {CCNPP Unit 3} demonstrate, under worst case maximum heat load conditions with an initial ESW cooling tower basin of 90°F, that the maximum ESW System design basis temperature of 95°F is not exceeded.

29. Safeguard Building or Fuel Building Boundary Inoperability

GTS Bases:

GTS Bases 3.7.12, "Safeguard Building Controlled Area Ventilation System (SBVS)," contains a Reviewer's Note that, in the Actions section for Required Action B.1, indicates that the adoption of Condition B is dependent on a commitment from the licensee to have guidance available describing compensatory measures to be taken in the event of intentional or unintentional entry into Condition B.

{CCNPP Unit 3} TS Bases:

{CCNPP Unit 3} Bases 3.7.12, "Safeguard Building Controlled Area Ventilation System (SBVS)," is revised, in the Actions section for Required Action B.1, to remove the Reviewer's Note and modify the Bases discussion for Required Action B.1 to include the required commitment.

Justification:

The site specific commitment provided is consistent with the requirements in the Reviewer's Note for adoption of the allowance provided in Condition B of TS 3.7.12, "Safeguard Building Controlled Area Ventilation System (SBVS)."

Part 4 Technical Specifications and Bases

Section B – {CCNPP Unit 3} Technical Specifications and Bases

A complete copy of the {CCNPP Unit 3} Technical Specifications and Bases are provided in this section. This copy incorporates the plant specific information and values, removes Reviewer's Notes and incorporates the departures addressed in Section A of this part of the COL Application.

1.1 Definitions

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status to other indications or status derived from independent instrumentation channels measuring the same parameter.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.3. Plant operation within these limits is addressed in individual Specifications.
DIVISION OPERATIONAL TEST (DOT)	A DOT shall be the injection of a simulated or actual signal into the division as close to the sensor as practicable to verify OPERABILITY of all devices in the division required for OPERABILITY. The DOT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for division OPERABILITY such that the setpoints are within the necessary range and accuracy. The DOT may be performed by means of any series of sequential, overlapping, or total steps.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

3.3 INSTRUMENTATION

3.3.1 Protection System (PS)

LCO 3.3.1 The PS sensors, manual actuation switches, signal processors, and actuation devices specified in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

-- NOTE --
Separate Condition entry is allowed for each sensor, manual actuation switch, signal processor, and actuation device.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more sensors inoperable.	A.1 --------------NOTE--------------- Only applicable for Table 3.3.1-1, Component A.21. -------------------------------------	
	Place inoperable sensor in trip.	1 hour
	AND	
	A.2 --------------NOTE--------------- Not applicable for Table 3.3.1-1, Component A.21. -------------------------------------	
	Place inoperable sensor in lockout.	4 hours
B. One or more manual actuation switches inoperable.	B.1 Restore manual actuation switch to OPERABLE status.	48 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more acquisition and processing units (APUs) inoperable due to the ~~Limiting Trip Setpoint (LTSP)~~Setpoint Control Program requirements for one or more Trip/Actuation Functions not met.	C.1	--------------NOTE--------------- Only applicable for APUs associated with Table 3.3.1-2, Trip/Actuation Functions B.10.a and B.10.b. -------------------------------------	
		Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown," for emergency diesel generator (EDG) made inoperable by inoperable APU.	1 hour
	AND		
	C.2	--------------NOTE--------------- Not applicable for APUs associated with Table 3.3.1-2, Trip/Actuation Functions B.10.a and B.10.b. -------------------------------------	
		Place the Trip/Actuation Function in the associated APU in lockout.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more signal processors inoperable for reasons other than Condition C.	D.1	--------------NOTE--------------- Only applicable for APUs associated with Table 3.3.1-2, Trip/Actuation Functions B.10.a and B.10.b. -------------------------------------	
		Enter applicable Conditions and Required Actions of LCO 3.8.1 and LCO 3.8.2 for EDG made inoperable by inoperable APU.	1 hour
	<u>AND</u>		
	D.2	--------------NOTE--------------- Not applicable for APUs associated with Table 3.3.1-2, Trip/Actuation Functions B.10.a and B.10.b. -------------------------------------	
		Place inoperable signal processor in lockout.	4 hours
E. One or more actuation devices inoperable.	E.1	Restore actuation device to OPERABLE status.	48 hours
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. <u>OR</u> Minimum functional capability specified in Table 3.3.1-1 not maintained.	F.1	Enter the Condition referenced in Table 3.3.1-1.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. As required by Required Action F.1 and referenced in Table 3.3.1-1.	G.1	Reduce THERMAL POWER to < 70% RTP.	2 hours
H. As required by Required Action F.1 and referenced in Table 3.3.1-1.	H.1	Reduce THERMAL POWER to < 10% RTP.	6 hours
I. As required by Required Action F.1 and referenced in Table 3.3.1-1.	I.1	Be in MODE 2.	6 hours
J. As required by Required Action F.1 and referenced in Table 3.3.1-1.	J.1	Be in MODE 3.	6 hours
K. As required by Required Action F.1 and referenced in Table 3.3.1-1.	K.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	K.2	Open the reactor trip breakers.	6 hours
L. As required by Required Action F.1 and referenced in Table 3.3.1-1.	L.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	L.2	Reduce pressurizer pressure to < 2005 psia.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
M. As required by Required Action F.1 and referenced in Table 3.3.1-1.	M.1	Be in MODE 3.	6 hours
	AND		
	M.2	Be in MODE 4.	12 hours
N. As required by Required Action F.1 and referenced in Table 3.3.1-1.	N.1	Be in MODE 3.	6 hours
	AND		
	N.2	Be in MODE 5.	36 hours
O. As required by Required Action F.1 and referenced in Table 3.3.1-1.	O.1	Declare associated EDG inoperable.	Immediately
P. As required by Required Action F.1 and referenced in Table 3.3.1-1.	P.1	Declare associated Chemical and Volume Control System isolation valve(s) inoperable.	Immediately
Q. As required by Required Action F.1 and referenced in Table 3.3.1-1.	Q.1	Declare associated Pressurizer Safety Relief Valve(s) inoperable.	Immediately
R. As required by Required Action F.1 and referenced in Table 3.3.1-1.	R.1	Declare both Control Room Emergency Filtration trains inoperable.	Immediately
S. As required by Required Action F.1 and referenced in Table 3.3.1-1.	S.1	Open reactor trip breakers.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
T. As required by Required Action F.1 and referenced in Table 3.3.1-1.	T.1 Declare associated Actuation Logic Units inoperable. AND T.2 Open reactor trip breakers.	Immediately 1 hour

SURVEILLANCE REQUIREMENTS

---NOTES---

1. Refer to Table 3.3.1-1 to determine which SRs apply for each sensor, manual actuation switch, signal processor, or actuation device.

2. When a sensor, manual actuation switch, signal processor, or actuation device is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Trip/Actuation Function maintains functional capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 20% RTP. --- Compare results of calorimetric heat balance calculation to power range division output. Adjust power range division output if calorimetric heat balance calculations results exceed power range division output by more than +2% RTP.	24 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	-------------------------------NOTE------------------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 20% RTP. -- Perform CALIBRATION.	15 effective full power days
SR 3.3.1.3	Perform ACTUATION DEVICE OPERATIONAL TEST.	31 days
SR 3.3.1.4	Perform CALIBRATION consistent with Specification 5.5.18, “Setpoint Control Program (SCP).”	92 days
SR 3.3.1.5	Perform a SENSOR OPERATIONAL TEST.	24 months
SR 3.3.1.6	------------------------NOTE-------------------------- Neutron detectors are excluded from CALIBRATION. -- Perform a CALIBRATION~~.~~ consistent with Specification 5.5.18, “Setpoint Control Program (SCP).”	24 months
SR 3.3.1.7	Perform EXTENDED SELF TESTS.	24 months
SR 3.3.1.8	Perform ACTUATION DEVICE OPERATIONAL TEST.	24 months
SR 3.3.1.9	Verify setpoints properly loaded in APUs.	24 months

Table 3.3.1-1 (page 1 of 3)
Protection System Sensors, Manual Actuation Switches,
Signal Processors, and Actuation Devices

COMPONENT	REQUIRED NUMBER OF SENSORS, SWITCHES, SIGNAL PROCESSORS, OR ACTUATION DEVICES	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUMUM REQUIRED FOR FUNCTIONAL CAPABILITY	CONDITION	SURVEILLANCE REQUIREMENTS
A. Sensors					
1. 6.9 kV Bus Voltage	3 per EDG	1,2,3,4,(a)	2 per EDG	O	SR 3.3.1.5 SR 3.3.1.6
2. Boron Concentration - Chemical and Volume Control System (CVCS) Charging Line	4	3[(b)],4[(b)],5,6	2	P	SR 3.3.1.4 SR 3.3.1.5
3. Boron Temperature - CVCS Charging Line	4	3[(b)],4[(b)],5,6	2	P	SR 3.3.1.5 SR 3.3.1.6
4. CVCS Charging Line Flow	4	3[(b)],4[(b)],5[(b)]	2	P	SR 3.3.1.5 SR 3.3.1.6
5. Cold Leg Temperature (Narrow Range)	4	≥ 10% RTP	3	H	SR 3.3.1.5 SR 3.3.1.6
6. Cold Leg Temperature (Wide Range)	4	1,2[(c)]	3	J	SR 3.3.1.5 SR 3.3.1.6
	4	3,4,5,6[(b)]	2	P	SR 3.3.1.5 SR 3.3.1.6
7. Containment Pressure	4 per area	1,2,3	3 per area	M	SR 3.3.1.5 SR 3.3.1.6
8. Hot Leg Pressure (Wide Range)	4	1,2,3	3	M	SR 3.3.1.5 SR 3.3.1.6
	4	(d)	2	Q	SR 3.3.1.5 SR 3.3.1.6
9. Hot Leg Temperature (Narrow Range)	4 per division, 4 divisions	1,2[(c)]	3 per division, 3 divisions	J	SR 3.3.1.5 SR 3.3.1.6
10. Hot Leg Temperature (Wide Range)	4	3[(e)]	3	M	SR 3.3.1.5 SR 3.3.1.6

(a) When associated EDG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

(b) With three or more reactor coolant pumps (RCPs) in operation.

(c) ≥ 10^{-5}% power on the intermediate range detectors.

(d) When Pressurizer Safety Relief Valves (PSRVs) are required to be OPERABLE per LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)."

(e) When Table 3.3.1-2, Trip/Actuation Function B.3.a is disabled.

Table 3.3.1-1 (page 2 of 3)
Protection System Sensors, Manual Actuation Switches,
Signal Processors, and Actuation Devices

COMPONENT	REQUIRED NUMBER OF SENSORS, SWITCHES, SIGNAL PROCESSORS, OR ACTUATION DEVICES	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUMUM REQUIRED FOR FUNCTIONAL CAPABILITY	CONDITION	SURVEILLANCE REQUIREMENTS
11. Intermediate Range	4	1[f],2,3[g]	3	K	SR 3.3.1.5 SR 3.3.1.6
12. Power Range	2 per division, 4 divisions	1,2,3[g]	2 per division, 3 divisions	K	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6
13. Pressurizer Level (Narrow Range)	4	1,2,3	3	M	SR 3.3.1.5 SR 3.3.1.6
14. Pressurizer Pressure (Narrow Range)	4	1,2,3[h]	3	L	SR 3.3.1.5 SR 3.3.1.6
15. Radiation Monitor - Containment High Range	4	1,2,3,4	3	K	SR 3.3.1.5 SR 3.3.1.6
16. Radiation Monitor - Control Room HVAC Intake Activity	4	1,2,3,4	3	N	SR 3.3.1.5 SR 3.3.1.6
	4	5,6,(i)	3	R	SR 3.3.1.5 SR 3.3.1.6
17. RCP Current	3 per RCP	1,2,3	2 per RCP	M	SR 3.3.1.5 SR 3.3.1.6
18. RCP Delta P Sensors	2 per RCP	1,2,3	1 per RCP	M	SR 3.3.1.5 SR 3.3.1.6
19. RCP Speed	4	≥ 10% RTP	3	H	SR 3.3.1.5 SR 3.3.1.6
20. Reactor Coolant System (RCS) Loop Flow	4 per loop	1,2[c]	3 per loop	J	SR 3.3.1.5 SR 3.3.1.6
21. Reactor Trip Circuit Breaker Position Indication	4	1,2[g],3[g]	3	M	SR 3.3.1.5 SR 3.3.1.8
22. Self-Powered Neutron Detectors	72	≥ 10% RTP	51	H	SR 3.3.1.2 SR 3.3.1.5

(c) ≥ 10^{-5} % power on the intermediate range detectors.

(f) ≤ 10% RTP.

(g) With the Reactor Control, Surveillance and Limitation (RCSL) System capable of withdrawing a Rod Cluster Control Assembly (RCCA) or one or more RCCAs not fully inserted.

(h) With pressurizer pressure ≥ 2005 psia.

(i) During movement of irradiated fuel assemblies.

Table 3.3.1-1 (page 3 of 3)
Protection System Sensors, Manual Actuation Switches,
Signal Processors, and Actuation Devices

COMPONENT	REQUIRED NUMBER OF SENSORS, SWITCHES, SIGNAL PROCESSORS, OR ACTUATION DEVICES	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUMUM REQUIRED FOR FUNCTIONAL CAPABILITY	CONDITION	SURVEILLANCE REQUIREMENTS
23. Steam Generator (SG) Level (Narrow Range)	4 per SG	1,2(j),3(j)	3 per SG	M	SR 3.3.1.5 SR 3.3.1.6
24. SG Level (Wide Range)	4 per SG	1,2,3	3 per SG	M	SR 3.3.1.5 SR 3.3.1.6
25. SG Pressure	4 per SG	1,2,3	3 per SG	M	SR 3.3.1.5 SR 3.3.1.6
B. Manual Actuation Switches					
1. Reactor Trip	4	1,2,3(g)	3	K	SR 3.3.1.8
	4	4(g),5(g)	3	S	SR 3.3.1.8
2. Safety Injection System (SIS) Actuation	4	1,2,3,4	3	N	SR 3.3.1.8
3. SG Isolation	4 per SG	1,2,3	3 per SG	M	SR 3.3.1.8
C. Signal Processors					
1. Remote Acquisition Units (RAUs)	2 per division, 4 divisions	≥ 10% RTP	1 per division, 4 divisions	H	SR 3.3.1.5 SR 3.3.1.7
2. Acquisition and Processing Units (APUs)	5 per division, 4 divisions	Refer to Table 3.3.1-2	Refer to Table 3.3.1-2	Refer to Table 3.3.1-2	SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.9
3. Actuation Logic Units (ALUs)	4 per division, 4 divisions	1,2,3,4	3 per division, 4 divisions	N	SR 3.3.1.5 SR 3.3.1.7
	4 per division, 4 divisions	5,6,(i)	3 per division, 4 divisions	T	SR 3.3.1.5 SR 3.3.1.7
D. Actuation Devices					
1. Reactor Coolant Pump Bus and Trip Breakers	2 per pump	1,2,3,4	1 per pump	N	SR 3.3.1.8
2. Reactor Trip Circuit Breakers	4	1,2,3(g)	3	K	SR 3.3.1.3
3. Reactor Trip Contactors	4 per set, 23 sets	1,2,3(g)	3 per set, 23 sets	K	SR 3.3.1.3

(g) With the RCSL capable of withdrawing a RCCA or one or more RCCAs not fully inserted.

(i) During movement of irradiated fuel assemblies.

(j) Except when all main feedwater (MFW) isolation valves are closed.

Table 3.3.1-2 (page 1 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

TRIP/ACTUATION FUNCTION		APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY(a)	SETTING BASIS~~LIMITING TRIP SETPOINT(b)(c)~~	CONDITION
A.	Reactor Trip				
1.a.	Low Departure from Nucleate Boiling Ratio (DNBR)	≥ 10% RTP	3 divisions	(b~~d~~)	H
1.b.	Low DNBR and Imbalance or Rod Drop	≥ 10% RTP	3 divisions	(b~~d~~)	H
1.c.	Variable Low DNBR and Rod Drop	≥ 10% RTP	3 divisions	(b~~d~~)	H
1.d.	Low DNBR - High Quality	≥ 10% RTP	3 divisions	(b~~d~~)	H
1.e.	Low DNBR - High Quality and Imbalance or Rod Drop	≥ 10% RTP	3 divisions	(b~~d~~)	H
2.	High Linear Power Density	≥ 10% RTP	3 divisions	(b~~d~~)	H
3.	High Neutron Flux Rate of Change (Power Range)	1,2,3(c~~e~~)	3 divisions	≥ 13~~1~~% RTP	K
4.	High Core Power Level	1,2(d~~f~~)	3 divisions	≤ 116.7~~05~~% RTP	J
5.	Low Saturation Margin	1,2(d~~f~~)	3 divisions	≥ 0~~[430]~~ Btu/lb	J
6.a.	Low-Low Reactor Coolant System (RCS) Loop Flow Rate in One Loop	≥ 70% RTP	3 divisions	≥ 50~~4~~% Nominal Flow	G
6.b.	Low RCS Loop Flow Rate in Two Loops	≥ 10% RTP	3 divisions	≥ 86~~90~~% Nominal Flow	H
7.	Low Reactor Coolant Pump (RCP) Speed	≥ 10% RTP	3 divisions	≥ 92~~3~~% Nominal Speed	H
8.	High Neutron Flux (Intermediate Range)	1(e~~g~~),2,3(c~~e~~)	3 divisions	≤ 2~~1~~5% RTP	K

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

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

~~(c) The setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the Trip/Actuation Function shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm Trip/Actuation Function performance. The methodologies used to determine the as-found and the as-left tolerances are specified in a document controlled under 10 CFR 50.59.~~

(b~~d~~) As specified in the COLR.

(c~~e~~) With the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted.

(d~~f~~) ≥ 10^{-5}% power on the intermediate range detectors.

(e~~g~~) ≤ 10% RTP.

Table 3.3.1-2 (page 2 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[(a)]	SETTING BASIS~~LIMITING TRIP SETPOINT[(b)(c)]~~	CONDITION
9. Low Doubling Time (Intermediate Range)	1[(e~~g~~)],2,3[(c~~e~~)]	3 divisions	≥ 1~~2~~0 Sec.	K
10. Low Pressurizer Pressure	≥ 10% RTP	3 divisions	≥ 1950~~2005~~ psia	H
11. High Pressurizer Pressure	1,2	3 divisions	≤ 2470~~15~~ psia	J
12. High Pressurizer Level	1,2	3 divisions	≤ 83~~75~~% Measuring Range	J
13. Low Hot Leg Pressure	1,2,3[(c~~e~~)(f~~h~~)]	3 divisions	≥ 1892~~200~~5 psia	L
14. Steam Generator (SG) Pressure Drop	1,2	3 divisions	≥ 29 psi/min; 177~~02~~ psi<ss; Max 1088 psia	J
15. Low SG Pressure	1,2,3[(c~~e~~)(f~~h~~)]	3 divisions	≥ 650~~725~~ psia	M
16. High SG Pressure	1	3 divisions	≤ 1460~~385~~ psia	I
17. Low SG Level	1,2	3 divisions	≥~~≤~~ 3.5~~20~~% Narrow Range	J
18. High SG Level	1,2	3 divisions	≤~~≥~~ 80.5~~69~~% Narrow Range ~~for 10 sec.~~	J
19. High Containment Pressure	1,2	3 divisions	≤ 1~~8.7~~9.2 psia	J

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

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

~~(c) The setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the Trip/Actuation Function shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm Trip/Actuation Function performance. The methodologies used to determine the as-found and the as-left tolerances are specified in a document controlled under 10 CFR 50.59.~~

(c~~e~~) With the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted.

(e~~g~~) ≤ 10% RTP.

(f~~h~~) With pressurizer pressure ≥ 2005 psia.

Table 3.3.1-2 (page 3 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

	TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[(a)]	SETTING BASIS~~LIMITING TRIP SETPOINT~~~~[(b)(c)]~~	CONDITION
B.	ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) SIGNALS				
1.	Turbine Trip on Reactor Trip (RT)	1	3 divisions	RT for 1 sec.	I
2.a.	Main Feedwater Full Load Closure on Reactor Trip (All SGs)	1,2[(g~~i~~)]	3 divisions	NA	J
2.b.	Main Feedwater Full Load Closure on High SG Level (Affected SGs)	1,2[(g~~i~~)],3[(g~~i~~)]	3 divisions	≤~~≥~~ 80.5~~69~~% Narrow Range for 10 sec.	M
2.c.	Startup and Shutdown Feedwater Isolation on SG Pressure Drop (All SGs)	1,2[(h~~j~~)],3[(h~~j~~)]	3 divisions	≥ 29 psi/min; 322~~247~~ psi<ss; Max 943 psia	M
2.d.	Startup and Shutdown Feedwater Isolation on Low SG Pressure (All SGs)	1,2[(h~~j~~)],3[(f~~h~~)(h~~j~~)]	3 divisions	≥ 505~~80~~ psia	L
2.e.	Startup and Shutdown Feedwater Isolation on High SG Level for Period of Time (Affected SGs)	1,2[(h~~j~~)],3[(h~~j~~)]	3 divisions	≤~~≥~~ 66.5~~69~~% Narrow Range for 10 sec.	M
3.a.	Safety Injection System (SIS) Actuation on Low Pressurizer Pressure	1,2,3[(j~~h~~)]	3 divisions	≥ ~~1668~~ 1613 psia	L
3.b.	SIS Actuation on Low Delta Psat	3[(i~~k~~)]	3 divisions	≥ ~~220~~ 39 psia	M
4.	RCP Trip on Low Delta P across RCP with SIS Actuation	1,2,3	3 divisions	≥ ~~80~~75% Nominal Pressure	M
5.	Partial Cooldown Actuation on SIS Actuation	1,2,3	3 divisions	NA	M

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

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

~~(c) The setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the Trip/Actuation Function shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm Trip/Actuation Function performance. The methodologies used to determine the as-found and the as-left tolerances are specified in a document controlled under 10 CFR 50.59.~~

(f~~h~~) With pressurizer pressure ≥ 2005 psia.

(g~~i~~) Except when all MFW full load isolation valves are closed.

(h~~j~~) Except when all MFW low load isolation valves are closed.

(i~~k~~) When Trip/Actuation Function B.3.a is disabled.

Table 3.3.1-2 (page 4 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

	TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[(a)]	SETTING BASIS~~LIMITING TRIP SETPOINT~~~~[(b)(c)]~~	CONDITION
6.a.	Emergency Feedwater System (EFWS) Actuation on Low-Low SG Level (All SGs)	1,2,3	3 divisions	≥ ~~40~~29% Wide Range	M
6.b.	EFWS Actuation on Loss of Offsite Power (LOOP) and SIS Actuation (All SGs)	1,2	3 divisions	NA	J
6.c.	EFWS Isolation on High SG Level (Affected SG)	1,2,3	3 divisions	≤ ~~89~~98% Wide Range	M
7.a.	Main Steam Relief Train (MSRT) Actuation on High SG Pressure	1,2,3	3 divisions	≤ 1460~~1385~~ psia	M
7.b.	MSRT Isolation on Low SG Pressure	1,2,3[(f~~h~~)]	3 divisions	≥ 505~~80~~ psia	L
8.a.	Main Steam Isolation Valve (MSIV) Closure on SG Pressure Drop (All SGs)	1,2,3	3 divisions	≥ 29 psi/min; ~~102~~ 177 psi<ss; Max 1088 psia	M
8.b.	MSIV Closure on Low SG Pressure (All SGs)	1,2,3[(j~~l~~)]	3 divisions	≥ 650~~725~~ psia	L
9.a.	Containment Isolation (Stage 1) on High Containment Pressure	1,2,3	3 divisions	≤ 1~~8.7~~9.2 psia	M
9.b.	Containment Isolation (Stage 1) on SIS Actuation	1,2,3,4	3 divisions	NA	N
9.c.	Containment Isolation (Stage 2) on High-High Containment Pressure	1,2,3	3 divisions	≤ ~~[~~38~~6~~.3~~]~~ psia	M
9.d.	Containment Isolation (Stage 1) on High Containment Radiation	1,2,3,4	3 divisions	≤ 100 x background	N

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

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

~~(c) The setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the Trip/Actuation Function shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm Trip/Actuation Function performance. The methodologies used to determine the as-found and the as-left tolerances are specified in a document controlled under 10 CFR 50.59.~~

(f~~h~~) With pressurizer pressure ≥ 2005 psia.

(j~~l~~) Except when all MSIVs are closed.

Table 3.3.1-2 (page 5 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[a]	SETTING BASIS~~LIMITING TRIP SETPOINT~~~~[(b)(c)]~~	CONDITION
10.a. Emergency Diesel Generator (EDG) Start on Degraded Grid Voltage	1,2,3,4,(k~~m~~)	NA	≥ ~~6210~~ 6089 V and ≤ ~~6350~~ 6486 V; ≥ ~~7~~ 6.5 sec. and ≤ ~~11~~ 12 sec. w/SIS, ≥ ~~270~~ 6.5 sec. and ≤ 300 sec. wo/SIS	NA
10.b. EDG Start on LOOP	1,2,3,4,(k~~m~~)	NA	≥ ~~4830~~ 4692 V and ≤ ~~4970~~ 5085 V; ≥ 0.~~4~~17 sec. and ≤ 7~~0.6~~ sec.	NA
11.a. Chemical and Volume Control System (CVCS) Charging Line Isolation on High-High Pressurizer Level	1,2,3	3 divisions	≤ ~~80~~88% Measuring Range	M
11.b. CVCS Charging Line Isolation on Anti-Dilution Mitigation (ADM) at Shutdown Condition (RCP not operating)	5[(l~~n~~)],6	3 divisions	~~927 ppm~~(b)	P
11.c. CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions	3,4[(~~o~~m)],5[(m~~o~~)]	3 divisions	(b~~d~~)	P

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

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

~~(c) The setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the Trip/Actuation Function shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm Trip/Actuation Function performance. The methodologies used to determine the as-found and the as-left tolerances are specified in a document controlled under 10 CFR 50.59.~~

(b~~d~~) As specified in the COLR.

(k~~m~~) When associated EDG is required to be OPERABLE by LCO 3.8.2.

(l~~n~~) With two or less RCPs in operation.

(m~~o~~) With three or more RCPs in operation.

Table 3.3.1-2 (page 6 of 6)
Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY[(a)]	SETTING BASIS~~LIMITING TRIP SETPOINT[(b)(c)]~~	CONDITION
12.a. Pressurizer Safety Relief Valve (PSRV) Actuation - First Valve	(n~~p~~)	3 divisions	(o~~q~~)	Q
12.b. PSRV Actuation - Second Valve	(n~~p~~)	3 divisions	(o~~q~~)	Q
13. Control Room Heating, Ventilation, and Air Conditioning Reconfiguration to Recirculation Mode on High Intake Activity	1,2,3,4	3 divisions	≤ 3 x background	N
	5,6,(p~~r~~)	3 divisions	≤ 3 x background	R

(a) A division is OPERABLE provided: a) the minimum sensors required for functional capability for all sensors providing input to the Trip/Actuation Function are OPERABLE; and b) the associated APU is OPERABLE.

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

~~(c) The setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the Trip/Actuation Function shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm Trip/Actuation Function performance. The methodologies used to determine the as-found and the as-left tolerances are specified in a document controlled under 10 CFR 50.59.~~

(n~~p~~) When the PSRVs are required to be OPERABLE by LCO 3.4.11.

(o~~q~~) The LTOP arming temperature is specified in the PTLR.

(p~~r~~) During movement of irradiated fuel assemblies.

~~[Reviewers Note: The values specified in brackets in the Limiting Trip Setpoint column are included for reviewer information only. A plant-specific setpoint study will be conducted. The values in Limiting Trip Setpoint column will then be replaced after the completion of this study.]~~

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - Shutdown

LCO 3.5.3 Two Medium Head Safety Injection (MHSI) trains shall be OPERABLE.

APPLICABILITY: MODE 4.

ACTIONS

---NOTE---
LCO 3.0.4.b is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required MHSI train inoperable.	A.1 Restore required MHSI train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two required MHSI trains inoperable.	B.1 Be in ~~Mode~~ <u>MODE</u> 5.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	The following SRs are applicable for all required MHSI trains: SR 3.5.2.2, SR 3.5.2.3, SR 3.5.2.4, SR 3.5.2.5, and SR 3.5.2.6.	In accordance with applicable SRs

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Filtration (CREF)

LCO 3.7.10 Two CREF trains shall be OPERABLE.

---NOTE---
The control room envelope (CRE) may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CREF train inoperable.	A.1	Restore CREF train to OPERABLE status.	7 days
B. Two CREF trains inoperable due to inoperable CRE boundary in MODE 1, 2, 3, or 4	B.1	Initiate action to implement mitigating actions.	Immediately
	AND		
	B.2	Verify mitigating actions ensure CRE occupant exposures to radiological~~,~~ ~~[[chemical]]~~, and smoke hazards will not exceed limits.	24 hours
	AND		
	B.3	Restore CRE boundary to OPERABLE status.	90 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	~~--------REVIEWER'S NOTE---------~~ ~~The need for the toxic gas isolation state will be determined by the COL Applicant.~~ ~~--~~ D.1 ~~-------------NOTE-------------~~ ~~Place CREF train in toxic gas isolation state if automatic transfer to toxic gas isolation state is inoperable.~~ ~~---------------------------------~~ Place OPERABLE CREF train in emergency mode. OR D.2 Suspend movement of irradiated fuel assemblies.	 Immediately Immediately
E. Two CREF trains inoperablein MODE 5 or 6, or during movement of irradiated fuel assemblies.	E.1 Suspend movement of irradiated fuel assemblies.	Immediately
F. Two CREF trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CREF train for ≥ 15 minutes with the heaters operating.	31 days
SR 3.7.10.2	Perform required CREF train testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.10.3	Verify each CREF train actuates on an actual or simulated actuation signal.	24 months
SR 3.7.10.4	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Storage Pool Boron Concentration

~~--Reviewer's Note--~~
~~The design of the spent fuel storage racks is to be provided by the COLA applicant. The required boron concentration will be provided as a part of the spent fuel rack design.~~
~~--~~

LCO 3.7.15 The spent fuel storage pool boron concentration shall be ≥ 500 ~~[1291]~~ ppm and boron enrichment shall be ≥ 37%.

APPLICABILITY: When fuel assemblies are stored in the spent fuel storage pool and a spent fuel storage pool verification has not been performed since the last movement of fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool boron concentration or enrichment not within limit.	------------------NOTE------------------ LCO 3.0.3 is not applicable. --	
	A.1 Suspend movement of fuel assemblies in the spent fuel storage pool.	Immediately
	AND	
	A.2.1 Initiate action to restore fuel storage pool boron concentration and enrichment to within limits.	Immediately
	OR	
	A.2.2 Initiate action to perform a spent fuel storage pool verification.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the spent fuel storage pool boron concentration is within limit.	7 days
SR 3.7.15.2	Verify the isotopic concentration of B^{10} in the spent fuel storage pool is within limit.	24 months

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Storage

LCO 3.7.16 ~~Fuel shall be stored in approved locations in the spent fuel storage pool~~The combination of initial enrichment and burnup of each fuel assembly stored in Region 2 of the spent fuel storage pool shall be within the limits specified in Figure 3.7.16-1.

~~---Reviewer's Note---~~
~~The design of the spent fuel storage racks is to be provided by the COLA applicant. The required spent fuel storage configuration will be provided as a part of the spent fuel rack design.~~
~~---~~

APPLICABILITY: Whenever any fuel assembly is stored in Region 2 of the spent fuel storage ~~racks~~pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	--------------NOTE-------------- LCO 3.0.3 is not applicable. ------------------------------------	
		Initiate action to ~~restore fuel storage to within requirements~~move the non-complying fuel assembly to an acceptable storage location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify by administrative means ~~that fuel assemblies are stored in approved locations~~the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.16-1.	Prior to storing the fuel assembly in Region 2 of the spent fuel storage

SURVEILLANCE	FREQUENCY
	pool

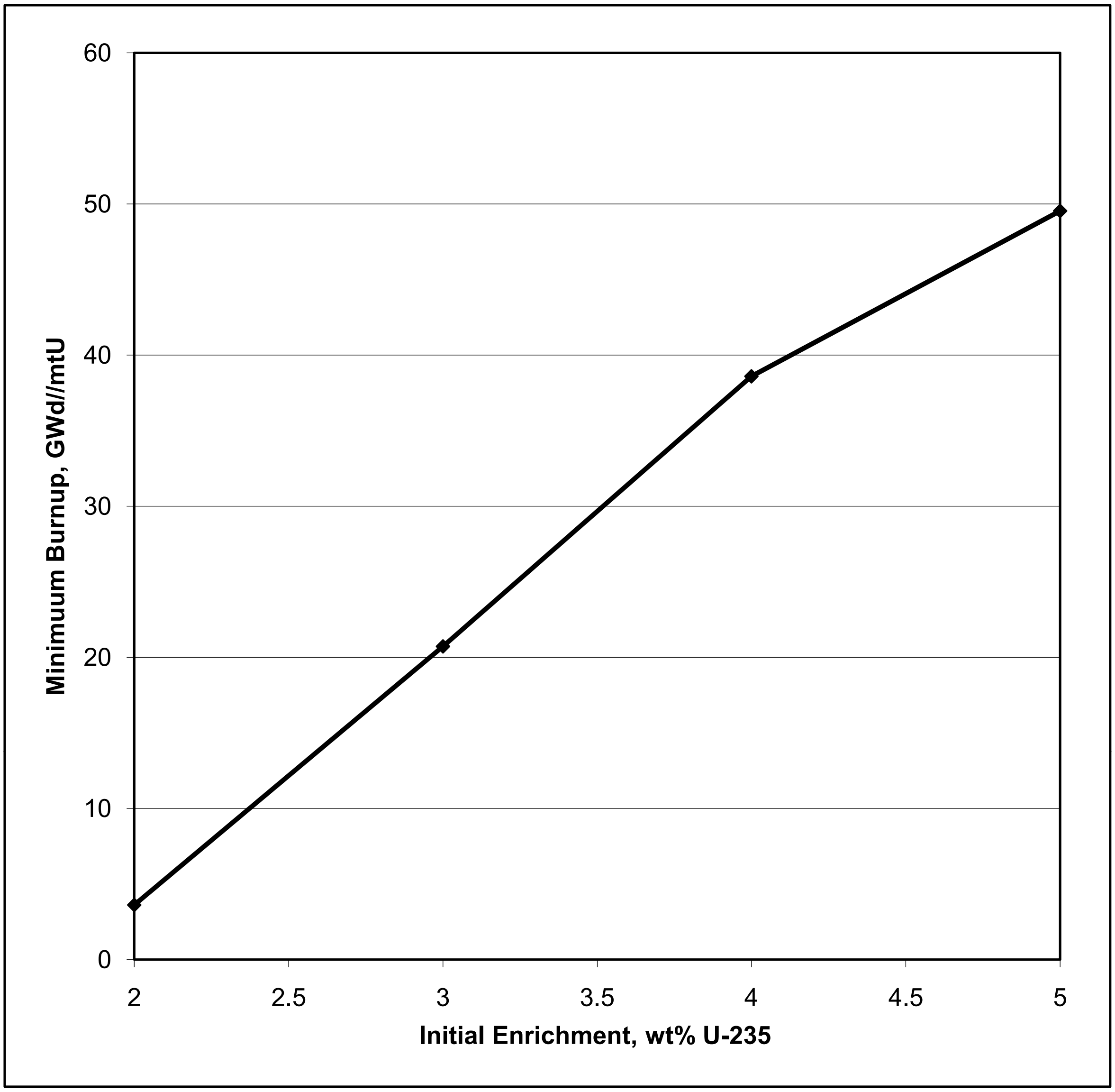


Figure 3.7.16-1 (page 1 of 1)

Fuel Assembly Burnup Requirements for Region 2

4.0 DESIGN FEATURES

4.1 Site Location

~~[A COL Applicant that references the U.S. EPR design certification will provide site-specific information for Section 4.1, Site Location.]~~The site for the Calvert Cliffs Nuclear Power Plant (CCNPP) Unit 3 is located on the western shore of the Chesapeake Bay in Calvert County, Maryland, about 10.5 miles southeast of Prince Frederick, Maryland. The site is approximately 45 miles southeast of Washington, DC, and 60 miles south of Baltimore, Maryland. The exclusion area boundary for CCNPP Unit 3 is a circle with a radius of 3324 feet. The exclusion area boundary establishes a radius of at least 2640 feet from potential CCNPP Unit 3 release points.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 241 fuel assemblies. Each assembly shall consist of a matrix of fuel rods clad with a zirconium based alloy with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 89 control rod assemblies. The control material shall be silver indium cadmium as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~[~~FSAR~~]~~ Section 9.1; ~~and~~

c. A nominal ~~[~~10~~1~~.9~~28]~~ inch center to center distance between fuel assemblies placed in ~~the spent fuel storage racks~~Region 1 and a nominal 9.028 inch center to center distance between fuel assemblies placed in Region 2 of the spent fuel storage racks;~~.~~

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

d. New or partially spent fuel assemblies with any discharge burnup may be allowed unrestricted storage in Region 1 of Figure 4.3-1;

e. Partially spent fuel assemblies meeting the initial enrichment and burnup requirements of LCO 3.7.16, “Spent Fuel Storage,” may be stored in Region 2 of Figure 4.3-1.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~[~~FSAR~~]~~ Section 9.1;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in ~~[~~FSAR~~]~~ Section 9.1; and

d. A nominal ~~[~~[10~~1~~.9~~]28]~~ inch center to center distance between fuel assemblies placed in the new fuel storage racks.

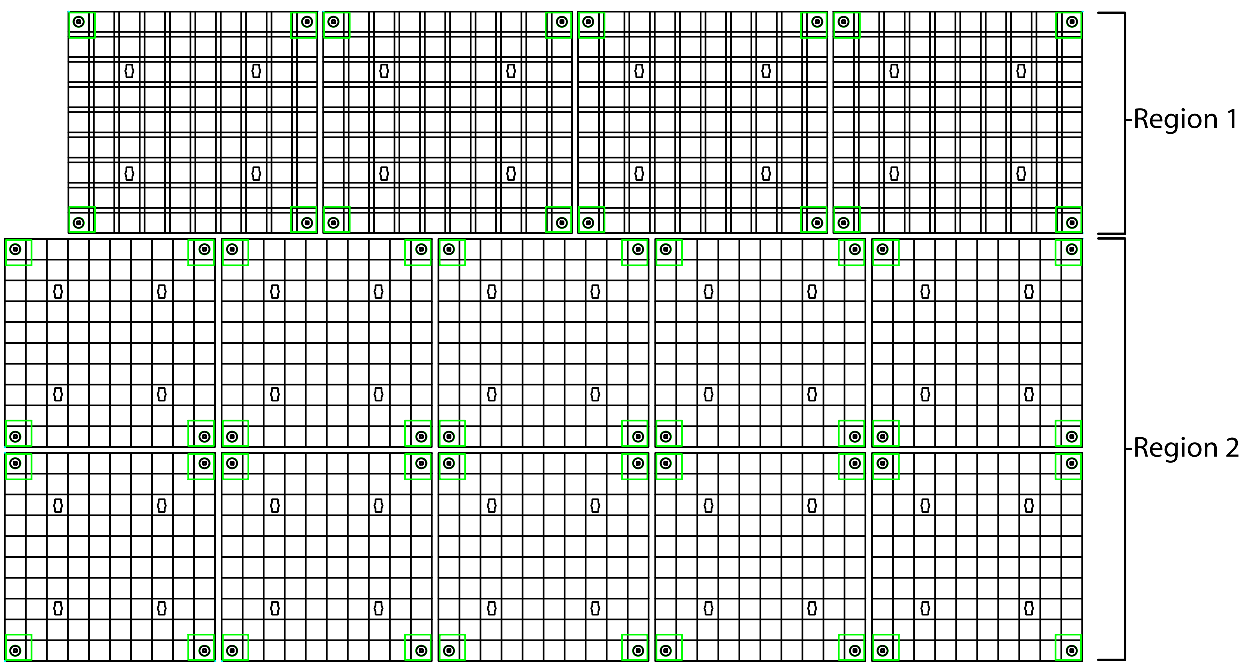
~~---REVIEWER’S NOTE---~~
~~Storage rack uncertainties are discussed in the FSAR or COLA Section 9.1~~
~~---~~

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 23 ft.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than ~~[1121~~1360~~]~~ fuel assemblies.



~~[~~Region 1 (Racks 1~~1~~ through ~~1~~4) – 360 locations

Region 2 (Racks 5~~1~~ through 14~~0~~) – 1000 locations

Total Storage Locations – 1360~~]~~

Figure 4.3-1 (page 1 of 1)

Discrete Two Region Spent Fuel Pool Rack Layout

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

--------------------------------------- ~~REVIEWER'S~~ NOTE ---------------------------------------

~~1.Titles for members of the unit staff shall be specified by use of an overall statement referencing an ANSI Standard acceptable to the NRC staff from which the titles were obtained, or an alternative title may be designated for this position. Generally, the first method is preferable; however, the second method is adoptable to those unit staffs requiring special titles because of unique organizational structures~~Organizational positions listed or described in the Administrative Controls section shall have corresponding plant-specific titles specified in the Final Safety Analysis Report.

~~1.The ANSI Standard shall be the same ANSI Standard referenced in Section 5.3, Unit Staff Qualifications. If alternative titles are used, all requirements of these Technical Specifications apply to the position with the alternative title applied with the specified title. Unit staff titles shall be specified in the Final Safety Analysis Report or Quality Assurance Plan. Unit staff titles shall be maintained and revised using those procedures approved for modifying/revising the Final Safety Analysis Report or Quality Assurance Plan.~~

5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

5.1.2 The shift supervisor shall be responsible for the control room command function. During any absence of the shift supervisor from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Operator license shall be designated to assume the control room command function. During any absence of the shift supervisor from the control room while the unit is in MODE 5 or 6, an individual with an active Senior Operator license or Operator license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the FSAR/QA Plan.

b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

The unit staff organization shall include the following:

~~------------------------------------REVIEWER'S NOTE------------------------------------~~
~~Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.~~
~~---~~

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODE 1, 2, 3, or 4;

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 5.2.2.a and 5.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements;

5.2 Organization

5.2.2 Unit Staff (continued)

c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position;

d. Administrative controls shall be developed and implemented to limit the working hours of personnel who perform safety related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel).

   The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

   Any deviation from the above guidelines shall be authorized in advance by the plant manager or designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized.

   Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned;

e. The operations manager or assistant operations manager shall hold a Senior Operator license; and

f. When the reactor is operating in MODE 1, 2, 3, or 4, an individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

~~---REVIEWER'S NOTE---~~
~~Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.~~
~~--~~

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 3, 2000, with the following exception:

a. During cold license operator training prior to Commercial Operation, the following Regulatory Position C.1.b of Regulatory Guide 1.8, Revision 2, 1987, applies:

Cold license operator candidates meet the training elements defined in ANS/ANSI 3.1-1993 but are exempt from the experience requirements defined in ANS/ANSI 3.1-1993.

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Operator and a licensed Operator are those individuals who, in addition to meeting the requirements of Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

5.5 Programs and Manuals

5.5.11 Gaseous Waste Processing System Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Waste Processing System and the quantity of radioactivity contained in gas delay beds. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure".

The program shall include:

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

b. A surveillance program to ensure that the quantity of radioactivity contained in the gas delay beds is less than the amount that would result in a whole body exposure of $\geq$ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the beds' contents.

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

~~--REVIEWER'S NOTE--~~
~~The U.S. EPR does not have outdoor liquid radwaste tanks. If a COL Applicant adds outdoor liquid radwaste tanks, this program will be modified accordingly.~~
~~--~~

5.5.12 Diesel Fuel Oil Testing Program

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

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or an absolute specific gravity within limits;

2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil; and

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

~~-------------------------------------REVIEWER'S NOTE-------------------------------------~~
~~As discussed in FSAR Section 6.2.6, the U.S. EPR has no penetrations that are classified as bypass leakage paths.~~
~~--~~

5.5.16 Battery Monitoring and Maintenance Program

This Program provides for battery restoration and maintenance, based on the recommendations of IEEE Standard 450-2002, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," or of the battery manufacturer including the following:

a. Actions to restore battery cells with float voltage < 2.13 V, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plate.

5.5.17 Control Room Envelope Habitability Program

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

a. The definition of the CRE and the CRE boundary;

b. Requirements for maintaining CRE boundary in its design condition including configuration control and preventive maintenance;

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

5.5 Programs and Manuals

5.5.17 Control Room Envelope Habitability Program (continued)

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization MODE of operation by one train of the CREFS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary;

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in Specification 5.5.17.c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences~~. Unfiltered air inleakage limits for [[hazardous chemicals]] must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis~~; and

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by Specifications 5.5.17.c and 5.5.17.d, respectively.

5.5.18 Setpoint Control Program (SCP)

a. The Setpoint Control Program shall document the Limiting Trip Setpoints (LTSPs), Nominal Trip Setpoints (NTSPs) (where desired), Allowable Values (AVs), and As-Found and As-Left Tolerance Bands for each of the required Technical Specification Instrumentation Functions in Specification 3.3.1, "Protection System (PS)."

b. The analytical methods used to determine the LTSPs, NTSPs (if applicable), AVs and As-Found Tolerance and As-Left Tolerance Bands shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. ANP-10275P-A, "U.S. EPR Instrument Setpoint Methodology Topical Report," ~~March 2007~~February 2008; and

2. ANP-10287, "Incore Trip Setpoint and Transient Methodology for U.S. EPR," November 2007.

5.5 Programs and Manuals

5.5.18 Setpoint Control Program (SCP) (continued)

c. The Setpoint Control Program shall also contain the following:

1. Provisions for evaluation of an instrumentation division to verify it is functioning as required, before return to service, if the as-found division setpoint is found to be conservative with respect to its AV, but outside its predefined As-Found Tolerance Band; and

2. Provisions for resetting an instrumentation division setpoint to a value that is within the As-Left Tolerance Band of the associated LTSP, or within the As-Left Tolerance Band of the associated NTSP (if applicable), or otherwise declaring the instrument division inoperable.

d. The Setpoint Control Program, including any revisions or supplements, shall be provided to the NRC upon issuance:

1. Prior to initial fuel load; and
2. On a frequency consistent with 10 CFR 50.71(e).

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Annual Radiological Environmental Operating Report

~~REVIEWER'S NOTE~~
~~[A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.]~~

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.2 Radioactive Effluent Release Report

~~REVIEWER'S NOTE~~
~~[A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.]~~

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

B 3.3 INSTRUMENTATION

B 3.3.1 Protection System (PS)

BASES

BACKGROUND

The PS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and breaching the reactor coolant pressure boundary during anticipated operational occurrences (AOOs). The PS also initiates the Engineered Safety Features (ESF) actuations that are used to mitigating accidents. The ESF actuates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits, maintain the Reactor Coolant System (RCS) pressure boundary, and mitigate the consequences of accidents that could result in potential exposures comparable to the guidelines set forth in 10 CFR 100 during AOOs and ensures acceptable consequences during accidents.

The PS initiates and the Safety Automation System (SAS) controls the necessary safety systems to protect against violating core design limits, maintain the RCS pressure boundary, and mitigate the consequences of accidents that could result in potential exposures comparable to the guidelines set forth in 10 CFR 100 during anticipated operational occurrences and ensures acceptable consequences during postulated accidents.

The four redundant divisions of the PS are physically separated in their respective safeguard buildings. The four divisionally separated rooms containing the PS equipment are in different fire zones. Therefore, in general, the consequences of internal hazards (e.g., fire), would impact only one PS division.

The PS architecture is four-fold redundant for both reactor trip and ESF functions. A single failure during corrective or periodic maintenance, or a single failure and the effects of an internal hazard does not prevent performance of the safety functions. For the reactor trip functions, each PS division actuates one division of the reactor trip devices based on redundant processing performed in four divisions. For ESF functions, the redundancy of the safety function as a whole is defined by the redundancy of the ESF system mechanical trains. In general, this results in one PS division actuating one mechanical train of an ESF system based on redundant processing performed in four divisions. The PS not only supports the redundancy of the mechanical trains, but also enhances this redundancy through techniques such as redundant actuation voting.

BASES

BACKGROUND (continued)

Three of the four divisions are necessary to meet the redundancy and testability of GDC 21 in 10 CFR 50, Appendix A (Ref. 3). The fourth division provides additional flexibility by allowing one division to be removed from service for maintenance or testing while still maintaining a minimum two-out-of-three logic. Thus, even with a division inoperable, no single additional failure in the PS can either cause an inadvertent trip/ESF or prevent a required trip/ESF from occurring.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the PS, as well as LCOs on other reactor system parameters and equipment performance. ~~The subset of LSSS that directly protect against violating the reactor core and RCS pressure boundary safety limits during AOOs are referred to as Safety Limit LSSS (SL-LSSS).~~

Technical Specifications are required by 10 CFR 50.36 to contain LSSS defined by the regulation as "...settings for automatic protective devices...so chosen 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 safety 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 protective devices 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. When LSSS is specified for a variable having a significant safety function, but which does not protect SLs, the setting must be chosen such that automatic protective actions will initiate consistent with the design basis. The Design Limit is the limit of the process variable at which the safety action is initiated to ensure that these automatic protective devices will perform their specified safety function. These limits (i.e., the Analytical Limits and Design Limits) constitute the Setting Basis specified in Table 3.3.1-2.

~~-------------------------------- REVIEWER'S NOTE -----------------------------------~~
~~The term "Limiting Trip Setpoint (LTSP)" is generic terminology for the setpoint value calculated by means of the plant-specific setpoint methodology documented in a document controlled under 10 CFR 50.59. The term LTSP indicates that no additional margin has been added between the Analytical Limit and the calculated trip setting. Where margin is added between the Analytical Limit and trip setpoint, the term~~

~~Nominal Trip Setpoint is preferred. The trip setpoint (field setting) may be more conservative than the Limiting or Nominal Trip Setpoint.~~

~~Where the LTSP is not included in Table 3.3.1-2 for the purpose of compliance with 10 CFR 50.36, the plant-specific term for the Limiting or Nominal Trip Setpoint must be cited in Note b of Table 3.3.1-2. The brackets indicate plant-specific terms may apply, as reviewed and approved by the NRC. The as-found and as-left tolerances will apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance.~~

BASES

BACKGROUND (continued)

~~Licensees are to insert the name of the document(s) controlled under 10 CFR 50.59 that contain the LTSP and the methodology for calculating the as-left and as-found tolerances, for the phrase "a document controlled under 10 CFR 50.59" in the specifications.~~

The LTSP is a predetermined setting for a protective device chosen to ensure automatic actuation prior to the process variable reaching the Analytical/Design Limit and thus ensuring that the SL would not be exceeded (i.e., for Analytical Limits) or that automatic protective actions occur consistent with the design basis (i.e., for Design Limits). As such, the LTSP accounts for uncertainties in setting the device (e.g., CALIBRATION), uncertainties in how the device might actually perform (e.g., repeatability), changes in the point of action of the device 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 LTSP ensures that SLs are not exceeded and that automatic protective devices will perform their specified safety function. ~~As such, the LTSP meets the definition of a SL-LSSS (Ref. 1).~~

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 function(s)." For automatic protective devices, the required safety function is to ensure that the SL is not exceeded and that automatic protective actions will initiate consistent with design basis. However, use of the LTSP 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 protective device 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 protective device with a setting that has been found to be different from the LTSP 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 LTSP and thus the automatic protective action would still have ensured that the SL would not be exceeded or that automatic protective actions would initiate consistent with the design basis with the "as-found" setting of the protective device. Therefore, the device would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the device to the ~~trip setpoint~~LTSP to account for further drift during the next surveillance interval.

~~However, there is also some point beyond which the device would have not been able to perform its function due, for example, to greater than~~

~~expected drift. This LTSP specified in Table 3.3.1-2 is the least conservative value of the as-found setpoint that a channel can have during testing such that a channel is OPERABLE if the trip setpoint is found conservative with respect to the Allowable Value during a SENSOR~~

BASES

BACKGROUND (continued)

~~OPERATIONAL TEST (SOT). As such, the Allowable Value differs from the LTSP by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the device will ensure that an SL is not exceeded at any given point of time as long as the device has not drifted beyond that expected during the surveillance interval. Note that, although the channel is OPERABLE under these circumstances, the setpoint must be left adjusted to a value within the as-left tolerance, and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found). If the actual setting of the device is found to be non-conservative with respect to the Allowable Value, the device would be considered inoperable from a Technical Specification perspective. This requires corrective action including those actions required by 10 CFR 50.36 when automatic protective devices do not function as required.~~ However, there is also some point beyond which the device would have not been able to perform its function due, for example, to greater than expected drift. This value is specified in the SCP, as required by Specification 5.5.18, in order to define OPERABILITY of the devices and is designated as the Allowable Value, which is the least conservative value of the as-found setpoint that a division can have during a periodic CALIBRATION or SENSOR OPERATIONAL TEST.

The actual LTSP and Allowable Values (derived for the Setting Basis values specified in Table 3.3.1-2) and the methodology for calculating the as-found and as-left tolerances are maintained in SCP, as required by Specification 5.5.18.

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

- The departure from nucleate boiling ratio (DNBR) shall be maintained above the SL value to prevent departure from nucleate boiling (DNB),

- Fuel centerline melting shall not occur; and

- The RCS pressure SL of 2803 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 100 (Ref. 2) criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within 10 CFR 100 limits. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The PS is segmented into four interconnected modules and associated LCOs for the reactor trips and ESF functions. These modules are:

- Sensors, which include the associated instrumentation;
- Manual actuation switches;

BASES

BACKGROUND (continued)

- Signal Processors, which include:
  - Remote Acquisition Units (RAUs), which acquire the signals from the Self-Powered Neutron Detectors (SPND) and distribute these signals;
  - Acquisition and Processing Units (APUs), which perform calculations and make setpoint comparisons; and
  - Actuation Logic Units (ALUs), which perform voting of the processing results from the redundant APUs in the different divisions and to issue actuation orders based on the voting results; and
- Actuation Devices, which includes the reactor trip breakers and contactors and the Priority Actuation and Control Systems (PACS) control modules for the Reactor Coolant Pump (RCP) bus and trip breakers..

The PS is a digital, integrated reactor protection system and engineered safety features actuation system. Individual sensors, signal processors, or the ALUs that provide the actuation signal voting function, can be associated with multiple reactor trip, ESF functions, and Permissives.

<u>Sensors</u>

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

The Power Density Detector System, which uses SPND and RAUs, provides the in-core monitoring function. The Power Range, Intermediate Range, and Source Range monitors provide the ex-core monitoring functions.

The instrument setpoint methodologies ~~used for the US EPR were submitted to NRC in References 1 and 4. The majority of PS trips or protection functions are based on single channel inputs; therefore, the uncertainties identified in Section 3.1 of Reference 1 are applicable for the trip. Reference 4 addresses the protection system trips or protection functions that are based on multiple inputs. The uncertainty calculations for the SPNDs, incore instrumentation, high linear power density, high~~<u>are discussed in the SCP.</u>

BASES

BACKGROUND (continued)

~~core power level, low saturation margin, anti-dilution, and DNBR use the statistical methodology described in Reference 4. As described therein, the LTSP is the LSSS since all known errors are appropriately combined in the total loop uncertainty calculation.~~

~~LTSPs in accordance with the Allowable Value will ensure~~The SCP ensures that appropriate settings are used for Trip/Actuation Functions and that SLs of Chapter 2.0, "Safety Limits (SLs)," are not violated during AOOs, and the consequences of postulated accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or postulated accident and the equipment functions as designed.

~~Note that the Allowable Values is the least conservative value of the as-found setpoint that a Trip/Actuation Function can have during a periodic CALIBRATION or SOT, such that a Trip/Actuation Function is OPERABLE if the as-found setpoint is conservative with respect to the Allowable Value.~~

Functional testing of the entire PS, from sensor input through the opening of individual sets of Reactor Trip Circuit Breakers (RTCB) or contactors, is performed each refueling cycle. Processing transmitter CALIBRATION is also normally performed on a refueling basis.

~~Trip Setpoints that directly protect against violating the reactor core or RCS pressure boundary Safety Limits during AOOs are SL-LSSS. Permissive setpoints allow bypass of trips when they are not required by the Safety Analysis. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventative 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, (i.e. the value indicated is sufficiently close to the necessary value to ensure proper operation of the safety systems to turn the AOO). Therefore permissives and interlocks are not considered to be SL-LSSS.~~

Manual Actuation Switches

Manual controls necessary to perform the manual operator actions credited in the safety analysis are included within the scope of the Technical Specifications. Manual actuation switches are provided to initiate the reactor trip function from the main control room (MCR) and the remote shutdown station (RSS). The ability to manually initiate ESF systems is provided in the MCR. Manual actuation of ESF systems initiates all actions performed by the corresponding automatic actuation including starting auxiliary or supporting systems and performing required sequencing functions.

Signal Processors

The PS is a distributed, redundant computer system. It consists of four independent redundant data-processing automatic paths (divisions), each with layers of operation and running asynchronous with respect to each other. In addition to the computers associated with the automatic paths, there are two redundant message and service interface computers to interface with each division.

The measurement channels or signal acquisition layer (which includes the RAUs) in each division acquires analog and binary input signals from sensors in the plant (such as for temperature, pressure, and level measurements). Each signal acquisition computer distributes its acquired and preprocessed input signals to the PS logic and controls, which includes the data processing computers (APUs).

BASES

BACKGROUND (continued)

The data-processing computers (APUs) perform signal processing for plant protective functions such as signal online validation, limit value monitoring and closed-loop control calculations. Each PS division contains four ALUs, two assigned to each subsystem. Two ALUs of the same subsystem within a division are redundant and perform the same processing using the same inputs. The outputs of two redundant ALUs are combined in a hardwired “functional AND” logic for reactor trip functions and in a hardwired OR logic for ESF functions. This avoids both unavailability of ESF functions and spurious reactor trips. The data processing computers then send their outputs to two independent voter computer units (ALUs) in each division.

In the voter computers, the outputs of the data-processing computers of redundant (three or four) divisions are processed together. A voter computer controls a set of actuators. Each voter receives the actuation signal from each of the redundant data-processing computers. The voter's task is to compare this redundant information and compute a validated (voted) actuating signal, which is used for actuating the end devices.

Each PS division contains four ALUs, two assigned to each subsystem. The two ALUs of the same subsystem within a division are redundant and perform the same processing using the same inputs. The outputs of two redundant ALUs are combined in a hardwired “functional AND” logic for reactor trip functions and in a hardwired OR logic for ESF functions.

For the reactor trip function, both ALUs in a division, if OPERABLE, must vote for an actuation. This provides protection against spurious trips. However, if only one ALU in a division is OPERABLE, the division is still OPERABLE, and the single voting ALU will initiate a reactor trip. For the ESF functions, an actuation will occur if either of the ALUs in a division votes for an actuation. This provides protection against ESF unavailability.

Reactor Trip Logic

Critical plant parameters such as temperatures, pressures, and levels are sensed, acquired, and converted to electrical signals by the PS. These signals are sent to various reactor trip functions in the PS where they are processed. When prohibited operating conditions exist, a reactor trip signal is generated from the reactor trip functions. Besides being generated automatically from the PS, a reactor trip signal can also be generated from the following systems:

BASES

BACKGROUND (continued)

- Automatic reactor trip from SAS in the event that the PS is lost;
- Manual trip from the Safety Information and Control System (SICS) panel. Four reactor trip switches are provided, which correspond to each of the four divisions;
- Manual trip from the control room; and
- Manual trip from the RSS. Note that the RSS manual trip is not part of the required circuits for LCO 3.3.1.

The reactor trip functions will utilize voting logic in order to screen out potential upstream failures of sensors or processing units. The architecture of the PS, as well as logic implemented in the system, will guard against spurious reactor trip orders while ensuring that those orders will be available when needed.

Single failures upstream of the ALU layer that could result in an invalid signal being used in the reactor trip actuation are marked as faulted by modifying the vote in the ALU layer. For the reactor trip functions, the vote is always modified toward actuation.

ESF Trip Logic

The ESF trip logic senses accident situations and initiate the operation of necessary features. The ESF along with reactor trip ensure the following:

- The integrity of the reactor coolant pressure boundary;
- The capability to shut down the reactor and maintain it in a safe shutdown condition; and
- The capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures.

BASES

BACKGROUND (continued)

As with the reactor trip logic, critical plant parameters such as temperatures, pressures, and levels are sensed, acquired, and converted to electrical signals by the PS. When prohibited operating conditions exist, an ESF signal is generated from the PS. In addition to the automatic ESF actuation functions performed by the PS, the capability to manually initiate these functions is provided in the MCR. These manual functions are implemented at the system level and perform the same actions as the automatic functions. The implementation of manual system level actuation of ESF functions and the priority between the automatic functions of the PS and the manual system level initiation is determined on a case-by-case basis.

Single failures upstream of the ALU layer that could result in an invalid signal being used in the ESF actuation are marked as faulted by modifying the vote in the ALU layer. For the ESF functions, the vote is modified toward actuation except:

- The Main Steam Relief Train (MSRT) divisions, which degrade towards isolation; and
- Pressurizer Safety Relief Valve (PSRV) opening for cold overpressure protection, which degrades towards non-actuation.

<u>Actuation Devices</u>

Reactor Trip Actuation Devices

The reactor trip actuation is performed by interrupting electrical power to the Control Rod Drive Mechanisms (CRMD). Electrical power to the CRDM is delivered by the Control Rod Drive Power Supply System (CRDPSS). The CRDPSS consists of 220 V DC distribution boards which are fed from the Uninterruptible Power Supply System.

BACKGROUND (continued)

The power supply of the CDRM can be switched off via the following features:

- Four main trip breakers distributed in two electrical divisions. Two breakers are located in Division 2, two others in Division 3. The main trip breakers can be opened by two coils: one with a de-energized logic using an under voltage coil and the other with an energized logic using a shunt trip coil.

- Four trip contactors combined in a 2-out-of-4 logic feed a group of four CRDM. Division 1, 2, and 3 contains eleven groups of four CRDMs. Division 4 contains eleven groups of four CRDMs and one single CRDM for the central rod. There are a total of 92 contactors. Each trip contactor is switched off by a de-energized coil.

- The electronics of the RodPilot can switch-off the power supply of four CRDMs. Two groups of four commands can actuate this electronic module, one with low active and one with high active logic. The electronics of the RodPilot is a non-safety device of the reactor trip but is the fastest switching device and allows the contactors and the trip breaker to open without stress.

- The under voltage coil of the main trip breakers is actuated by the automatic reactor trip signals of the PS and the manual trip from the SICS panel. The shunt coil of the main trip breakers is actuated by the automatic reactor trip signal from the SAS and the manual trip signal from the RSS. The shunt coil of the trip breakers receives two different signals from SAS and RSS combined in an "OR" logic performed at the level of trip breakers.

The operator can manually close the breakers by individual controls. This control actuates the closing coil of the breaker via the SAS. In the electronics of the breaker, the opening of trip breaker must have priority to the closing.

The reactor trip signal generated automatically by the PS and the manual trip signal generated from the SICS panel can actuate the trip contactors.

Engineered Safety Features Actuation Devices

The ESF determines the need for actuation in each of the input divisions monitoring each actuation parameter. Once the need for actuation is determined, the condition is transmitted to automatic actuation output logic divisions, which perform the logic to determine the actuation of each end device. Each end device has its own automatic actuation logic.

BACKGROUND (continued)

Each of the PS sensors, signal processors, or actuation devices can be placed in lockout, which renders the component inoperable. The digital signals within the PS carry a value and a status. The signal status can be propagated through the software function blocks; therefore, if an input signal to a function block has a faulty status, the output of the function block also has a faulty status. When a signal with a faulty status reaches the voting function block, the signal is disregarded through modification of the voting logic. Individual function computers can be put into a testing and diagnostic mode via the service unit. The function processor that is being tested then behaves like a computer with a "detected fault" for the system. The signal outputs are disabled and those sent via the communication means are marked with the status "TEST" or "ERROR" and therefore masked by selection blocks with active status processing. In this case the receiving function processor behaves as if the transmitting function processor had failed.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The PS is designed to ensure that the following operational criteria are met:

– The associated actuation will occur when the parameter monitored by each division reaches its setpoint and the specific coincidence logic is satisfied; and

– Separation and redundancy are maintained to permit a division to be out of service for testing or maintenance while still maintaining redundancy within the PS instrumentation network.

Each of the analyzed transients and accidents can be detected by one or more PS Functions. Each of the PS reactor trip and ESF Functions included in the Technical Specifications are credited as part of the primary success path in the accident analysis. Non-credited functions are purely equipment protective, and their use minimizes the potential for equipment damage. Non-credited functions are not included in the Technical Specifications. Refer to FSAR Sections 7.2 and 7.3.

The LCO requires the PS sensors, manual actuation switches, signal processors, and specified actuation devices to be OPERABLE. The LCO ensures that each of the following requirements is met:

– A reactor trip or ESF function will be initiated when necessary; and

– Sufficient redundancy is maintained to permit a component to be out of service for testing or maintenance.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Failure of any sensors, signal processors, or actuation device reduces redundancy or renders the affected division(s) inoperable.

~~Trip Setpoints that directly protect against violating the reactor core or RCS pressure boundary SLs during AOOs are SL-LSSS. Permissive and interlock setpoints allow bypass of trips when they are not required by the Safety Analysis. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventative 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, (i.e. the value indicated is sufficiently close to the necessary value to ensure proper operation of the safety systems to turn the AOO). Therefore permissives and interlocks are not considered to be SL-LSSS. Each LTSP specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip Function. The methodologies for considering uncertainties are defined in References 1 and 4.~~The Limiting Trip Setpoints, Allowable Values, and as-left and as-found tolerances, and the methodologies to calculate these values are specified in the SCP (Specification 5.5.18).

The PS sensors, manual actuation switches, signal processors, and specified actuation devices satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii) .

The PS sensors, manual actuation switches, signal processors, and specified actuation devices that support reactor trips are required to be OPERABLE in MODES 1, 2 and/or 3 because the reactor is or can be made critical in these MODES. The automatic reactor trip functions are designed to take the reactor subcritical, which maintains the SLs during AOOs and assists the ESF in providing acceptable consequences during accidents. The PS sensors, manual actuation switches, signal processors, and specified actuation devices that support automatic reactor trip functions are not required to be OPERABLE in MODES 4 and 5. In MODES 4 and 5, the emphasis is placed on return to power events. The reactor is protected in these MODES by ensuring adequate SDM.

The PS sensors, manual actuation switches, signal processors, and specified actuation devices that support reactor trips are required to be OPERABLE in MODES 1, 2, 3 and/or 4 since there is sufficient energy in the primary and secondary systems to warrant automatic ESF System responses to:

- Close the MSIVs to preclude a positive reactivity addition,

- Actuate Emergency Feedwater (EFW) to preclude the loss of the SGs as a heat sink (in the event the normal feedwater system is not available),

- Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by isolating containment and limiting the containment pressure from exceeding the containment design pressure during a design basis Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB), and

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

- Actuate ESF systems to ensure sufficient borated inventory to permit adequate core cooling and reactivity control during a design basis LOCA or MSLB accident.

In MODES 5 and 6, automatic actuation of the ESF Functions is not normally required because adequate time is available to evaluate plant conditions and respond by manually operating the ESF components if required. Exceptions to this are:

- ESF 10.a - Emergency Diesel Generator (EDG) Start on Degraded Grid Voltage,
- ESF 10.b - EDG Start on Loss of Offsite Power (LOOP),
- ESF 11.b - Chemical and Volume Control System (CVCS) Charging Line Isolation on Anti-Dilution Mitigation (ADM) at Shutdown Condition (RCP not operating),
- ESF 11.c - CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions,
- ESF 12.a and 12.b - PSRV Actuation - First and Second Valve, and
- ESF 13 - Control Room Heating, Ventilation and Air Conditioning (HVAC) Reconfiguration to Recirculation Mode on High Intake Activity.

These ESF functions are required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies to ensure that:

- Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;
- Systems needed to mitigate a fuel handling accident are available; and
- Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The specific safety analysis and OPERABILITY requirements applicable to each PS protective function is identified below.

A. REACTOR TRIPS

1. Low DNBR (Includes High Outlet Quality)

This function protects the fuel against the risk of departure from nucleate boiling during AOOs that lead to a decrease of the DNBR value. There are five Low DNBR trips:

a. Low DNBR,
b. Low DNBR and Imbalance or Rod Drop,
c. Variable Low DNBR and Rod Drop,
d. Low DNBR - High Quality, and
e. Low DNBR - High Quality and Imbalance or Rod Drop.

Together, these five trips protect against the following AOOs:

- Increase in heat removal by the secondary system,
- Decrease in heat removal by the secondary system,
- Reactivity and power distribution anomalies, and
- Decrease in reactor coolant inventory.

The Low DNBR (1.a) and High Quality (1.d) trips require four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- SPNDs,
- RCP speed sensor,
- Pressurizer Pressure (Narrow Range) sensor,
- Cold leg temperature (Narrow Range) sensor,
- RCS loop flow sensors,
- RAU,
- APUs, and
- ALUs.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Low DNBR and Imbalance or Rod Drop (1.b), Variable Low DNBR and Rod Drop (1.c), and High Quality and Imbalance or Rod Drop (1.e) trips require four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- SPNDs,
- Rod Cluster Control Assembly (RCCA) position indicators,
- RCP speed sensor,
- Pressurizer Pressure (Narrow Range) sensor,
- Cold leg temperature (Narrow Range) sensor,
- RCS loop flow sensors,
- RAU,
- RCCA Unit,
- APU, and
- ALUs.

The ~~LTSPs~~ Analytical Limits are high enough to provide an operating envelope that prevents an unnecessary low DNBR reactor trip. The Analytical Limits~~LTSPs~~ are low enough for the system to maintain a margin to unacceptable fuel cladding damage for AOOs that leads to an uncontrolled decrease of the DNBR value.

The P2 permissive automatically enables the five Low DNBR Trip signals when the neutron flux, as measured by the power range, is greater than or equal to 10% RTP. When nuclear power is below this threshold, the trips are also automatically disabled by Permissive P2.

2. High Linear Power Density

This function protects the fuel against the risk of melting at the center of the fuel pellet, during accidental transients, for events leading to an uncontrolled increase of the linear power density.

This trip protects against the following postulated accidents or AOOs:

- Increase in heat removal by the secondary system, and
- Reactivity and power distribution anomalies.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The High Linear Power Density Trip requires four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- SPNDs,
- RAU,
- APUs, and
- ALUs.

The Analytical Limits~~LTSPs~~ are high enough to provide an operating envelope that prevents unnecessary High Linear Power reactor trips. The Analytical Limits~~LTSPs~~ are low enough for the system to maintain a margin to unacceptable fuel centerline melt for any AOOs that lead to an uncontrolled increase of the linear power density.

The P2 permissive automatically enables the Reactor Trip signal when the neutron flux, as measured by the power range, is greater than or equal to 10% RTP. When nuclear power is below this threshold, the trip is also automatically disabled by Permissive P2.

3. High Neutron Flux Rate of Change (Power Range)

This function limits the consequences of an excessive reactivity increase from an intermediate power level including nominal power. This trip protects against reactivity and power distribution anomalies.

The High Neutron Flux Rate of Change Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and MODE 3 with the Reactor Control, Surveillance and Limitation (RCSL) System capable of withdrawing a RCCA or one or more RCCAs not fully inserted:

- Power Range sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents unnecessary Excore High Neutron Flux Rate of Change reactor trips. The Analytical Limit~~LTSP~~ is low enough for the system to maintain a margin to unacceptable fuel cladding damage due to an excessive reactivity increase from an intermediate power level including nominal power.

There are no permissives associated with this trip.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4. High Core Power Level

This function limits the consequences of an excessive reactivity increase from an intermediate high power level including nominal power. This trip protects against the following postulated accidents or AOOs:

- Increase in heat removal by the secondary system, and
- Reactivity and power distribution anomalies.

The High Core Power Level Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and in MODE 2 when the nuclear power level is greater than or equal to 10^{-5}% power as indicated on the Intermediate Range monitors:

- Cold Leg Temperature sensors (Wide Range),
- Hot Leg Temperature (Narrow Range) sensors,
- Hot Leg Pressure (Wide Range) sensors,
- RCS Loop Flow sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents an unnecessary High Core Power Level reactor trip. The Analytical Limit~~LTSP~~ is low enough for the system to maintain a margin to unacceptable fuel cladding damage due to an excessive reactivity increase from an intermediate high power level including nominal power.

The P5 permissive automatically enables the High Core Power Level Trip when the nuclear power level is greater than or equal to 10^{-5}% power. The P5 permissive also automatically disables the High Core Power Level Trip below this power.

5. Low Saturation Margin

This function limits the consequences of an excessive reactivity increase from an intermediate high power level including nominal power. This trip protects against the following postulated accidents or AOOs:

- Increase in heat removal by the secondary system, and
- Reactivity and power distribution anomalies.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Low Saturation Margin Trip requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 and MODE 2 when the nuclear power level is greater than or equal to 10^{-5}% power as indicated on the Intermediate Range monitors.:

- Cold Leg Temperature sensors (Wide Range),
- Hot Leg Temperature (Narrow Range) sensors,
- Hot Leg Pressure (Wide Range) sensors,
- RCS Loop Flow sensors,
- APUs, and
- ALUs.

The ~~LTSP~~ Design Limit is high enough to provide an operating envelope that prevents an unnecessary Low Saturation Margin reactor trip. The ~~LTSP~~ Design Limit is low enough for the system to maintain a margin to unacceptable fuel cladding damage during AOOs.

The P5 permissive automatically enables the Low Saturation Margin Trip when the nuclear power level is greater than or equal to 10^{-5}%. The P5 permissive also automatically disables the Low Saturation Margin Trip below this power.

6. RCS Loop Flow Rate

This function initiates a reactor trip and is inhibited below a certain level of nuclear power under which the protection is not necessary because DNB is no longer a risk in this condition. There are two trips:

a. Low-Low RCS Loop Flow Rate in One Loop, and
b. Low RCS Loop Flow Rate in Two Loops.

These trips protect against the following postulated accidents or AOOs:

- Decrease in heat removal by the secondary system, and
- Decrease in RCS flow rate.

The Low-Low RCS Loop Flow in One Loop Trip (6.a) requires four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 70% RTP:

- RCS Loop Flow sensors,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents unnecessary Low-Low Loop Flow Rate reactor trips. The Analytical Limit~~LTSP~~ is low enough for the system to maintain a margin to ensure DNBR limits are met for AOOs and bounded for postulated accidents.

The P3 permissive automatically enables the Low-Low RCS Loop Flow Rate Trip (One Loop) when the nuclear power level is greater than or equal to 70% RTP. The P3 permissive also automatically disables the Low-Low RCS Loop Flow Rate Trip (One Loop) below this power.

The Low RCS Loop Flow in Two Loops Trip (6.b) requires four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- RCS Loop Flow sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents unnecessary Low Loop Flow Rate reactor trips. The Analytical Limit~~LTSP~~ is low enough for the system to maintain a margin to ensure DNBR limits are met for AOOs.

The P2 permissive automatically enables the Low RCS Loop Flow Rate Trip (Two Loops) when the nuclear power level is greater than or equal to 10% RTP. The P2 permissive also automatically disables the Low RCS Loop Flow Rate Trip (Two Loops) when the nuclear power level is below this power.

7. Low RCP Speed

Due to electrical transients that may affect the RCP's, a specific protection function is required. This function initiates a reactor trip and is inhibited below a low level of reactor power under which the protection is not necessary because DNB is no longer a risk.

This trip protects against the following postulated accidents or AOOs:

- Decrease in heat removal by the secondary system, and
- Decrease in RCS flow rate.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Low RCP Speed Trip requires four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- RCP Speed Trip sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents unnecessary Low RCP Speed reactor trips. The Analytical Limit~~LTSP~~ is low enough for the system to maintain a margin to ensure DNBR limits are met for AOOs.

The P2 permissive automatically enables the Low RCP Speed Trip when the power level is greater than or equal to 10% RTP. When nuclear power is below this threshold, the trip is also automatically disabled by permissive function P2.

8. High Neutron Flux (Intermediate Range)

This function limits the consequences of an excessive reactivity increase when the reactor is started up from a sub-critical or low power start-up condition. This trip protects against reactivity and power distribution anomalies.

The High Neutron Flux Trip requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 when RTP is less than or equal to 10%, MODE 2, and in MODE 3 when RCSL is capable of withdrawing a RCCA or one or more RCCAs not fully inserted:

- Intermediate Range sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents an unnecessary High Neutron Flux reactor trip. The Analytical Limit ~~LTSP~~ is low enough for the system to maintain a margin to unacceptable fuel cladding damage for AOOs that leads to an uncontrolled increase of the linear power density.

The P6 permissive automatically enables the High Neutron Flux Intermediate Range reactor trip when the power level is less than or equal to 10% RTP.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

9. Low Doubling Time (Intermediate Range)

This function limits the consequences of an excessive reactivity increase when the reactor is started up from a sub-critical or low power start-up condition. This trip protects against reactivity and power distribution anomalies.

The Low Doubling Time Trip requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 when RTP is less than or equal to 10%, MODE 2, and in MODE 3 when RCSL is capable of withdrawing a RCCA or one or more RCCAs not fully inserted:

- Intermediate Range sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents an unnecessary Low Doubling Time reactor trip. The Analytical Limit~~LTSP~~ is low enough for the system to maintain a margin to unacceptable fuel cladding damage for any postulated event that leads to an uncontrolled increase of the linear power density.

The P6 permissive automatically enables the Low Doubling Time reactor trip when the power level is less than or equal to 10% RTP.

10. Low Pressurizer Pressure

A RCS depressurization may lead to a risk of excessive boiling, thus a reactor trip is required to ensure fuel rod integrity and to adapt reactor power to the capacity of the safety systems. This trip protects against a decrease in reactor coolant inventory.

The Low Pressurizer Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE when the reactor power level is greater than or equal to 10% RTP:

- Pressurizer Pressure (Narrow Range) sensors,
- APUs, and
- ALUs.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

A RCS depressurization may lead to a risk of excessive boiling, thus a reactor trip is required to ensure fuel rod integrity and to adapt reactor power to the capacity of the safety systems. The Analytical Limit~~LTSP~~ is sufficiently below the full load operating value for RCS pressure so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of an RCS depressurization.

The P2 permissive automatically enables the Low Pressurizer Pressure Trip when the power level is greater than or equal to 10% RTP. When nuclear power is below this threshold, the trip is automatically disabled by permissive function P2.

11. High Pressurizer Pressure

In case of a RCS overpressure, a reactor trip is required in order to:

- Adapt the reactor power to the capacity of the safety systems;
- Ensure RCS integrity; and
- Avoid opening of the Pressurizer safety valves in certain primary side overpressure analyses.

This trip protects against a decrease in heat removal by the secondary system.

The High Pressurizer Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2:

- Three Pressurizer Pressure (Narrow Range) sensors,
- Three divisions of APUs, and
- Three divisions of ALUs.

The Analytical Limit~~LTSP~~ is ~~set~~ below the nominal lift setting of the Pressurizer code safety valves, and its operation avoids the undesirable operation of these valves during normal plant operation. In the event of a complete loss of electrical load from 100% power, this setpoint ensures the reactor trip will take place, thereby limiting further heat input to the RCS and consequent pressure rise. The PSRVs may lift to prevent over pressurization of the RCS.

There are no permissives associated with this trip.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

12. High Pressurizer Level

In case of increasing Pressurizer level, a reactor trip is required in order to avoid Pressurizer over filling and to prevent the PSRVs from relieving. This trip protects against increases in reactor coolant inventory.

The High Pressurizer Level Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2:

- Pressurizer Level (Narrow Range) sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is ~~set~~ below the point where the associated transient would reach the nominal lift setting of the PSRVs, and its operation avoids the undesirable operation of these valves during normal plant operation. In the event of a CVCS malfunction, this Analytical Limit~~setpoint~~ ensures a timely reactor trip will take place in order to avoid filing the pressurizer. The PSRVs may lift to prevent over pressurization of the RCS.

The P12 permissive automatically enables the High Pressurizer Level Trip when the pressure is greater than or equal to 2005 psia.

13. Low Hot Leg Pressure

A RCS depressurization may lead to a risk of excessive boiling, thus a reactor trip is required to ensure fuel rod integrity and to adapt reactor power to the capacity of the safety systems. This trip protects against a decrease in reactor coolant inventory.

The Low Hot Leg Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and in MODE 3 with the pressurizer pressure greater than or equal to 2005 psia, when the RCSL System is capable of withdrawing a RCCA, or one or more RCCAs are not fully inserted.

- Hot Leg Pressure (Wide Range) sensors,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

A RCS depressurization may lead to a risk of excessive boiling, thus a reactor trip is required to ensure fuel rod integrity and to adapt reactor power to the capacity of the safety systems. The Analytical Limit~~LTSP~~ is sufficiently below the full load operating value so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of abnormal conditions.

The P12 permissive automatically enables the Low Hot Leg Pressure Trip when the pressure is greater than or equal to 2005 psia.

14. Steam Generator Pressure Drop

In case of steam or feedwater system piping failure, the affected Steam Generator (SG) depressurizes leading to a RCS cooldown and hence a reactivity transient. A reactor trip is required in order to ensure the fuel rod integrity and to adapt the reactor power to the capacity of the safety systems. This trip protects against the following postulated accidents or AOOs:

- Increase in heat removal by the secondary system, and
- Decrease in heat removal by the secondary system.

The SG Pressure Drop Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2:

- SG Pressure sensors,
- APUs, and
- ALUs.

In case of steam or feedwater system piping failure, the affected SG depressurizes leading to a RCS cooldown or heatup. A reactor trip is required in order to ensure the fuel rod integrity and to adapt the reactor power to the capacity of the safety systems. The Analytical Limit~~LTSP~~ is sufficiently below the full load operating value so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of a pipe break.

There are no permissives associated with this trip.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

15. Low SG Pressure

In case of steam or feedwater system piping failure, the affected SG depressurizes leading to a RCS cooldown and hence a criticality transient. For small breaks, the setpoint of the reactor trip on SG pressure drop may not be reached. Therefore, a reactor trip on low SG pressure is introduced in order to ensure fuel rod integrity and to adapt the reactor power to the capacity of safety systems. This trip protects against the following postulated accidents or AOOs:

- Increase in heat removal by the secondary system, and
- Decrease in heat removal by the secondary system.

The Low SG Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and in MODE 3 with either the pressurizer pressure greater than or equal to 2005 psia, the RCSL System capable of withdrawing a RCCA, or one or more RCCAs not fully inserted:

- SG Pressure sensors,
- APUs, and
- ALUs.

In case of steam or feedwater system piping failure, the affected SG depressurizes leading to a RCS cooldown or heatup. For small breaks, the setpoint of the reactor trip on SG pressure drop may not be reached. Therefore, a reactor trip on low SG pressure is introduced in order to ensure fuel rod integrity and to adapt the reactor power to the capacity of safety systems. The Analytical Limit~~LTSP~~ is sufficiently below the full load operating value so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of a pipe break.

The P12 permissive automatically enables the Low SG Pressure Trip when the pressure is greater than or equal to 2005 psia.

16. High SG Pressure

In case of a loss of the main heat sink, the reactor has to be tripped in order to:

- Ensure fuel rods integrity at power;
- Adapt the reactor power to the capacity if safety systems; and
- Ensure SG integrity.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This trip protects against a decrease in heat removal by the secondary system.

The High SG Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE in MODE 1:

- SG Pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is ~~set~~ high enough to avoid spurious operation. In case of a loss of the main heat sink, the Analytical Limit~~setpoint~~ is ~~set~~ low enough to trip the reactor in order to:

- Ensure fuel rod integrity at power,
- Adapt the reactor power to the capacity of safety systems, and
- Ensure SG integrity.

There are no permissives associated with this trip.

17. Low SG Level

This trip protects the reactor from a loss of heat sink in case of SG steam/feedwater flow mismatch. This trip protects against a decrease in heat removal by the secondary system.

The Low SG Level Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2:

- SG Level (Narrow Range) sensors,
- APUs, and
- ALUs.

The purpose of this trip is to protect the reactor from a loss of heat sink in case of SG steam/feedwater flow mismatch. The Analytical Limit~~LTSP~~ is sufficiently below the full load operating value so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of a flow mismatch.

The P13 permissive automatically enables the Low SG Level Trip when the hot leg temperature is greater than or equal to 200ºF.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

18. High SG Level

This trip protects the turbine against an excessive humidity in case of a Main Feedwater (MFW) malfunction causing an increase in feedwater flow or in case of SG level increase. This reactor trip ensures core integrity during these transients since an increase in feedwater flow leads to a RCS overcooling event and hence a reactivity insertion. This trip protects against an increase in heat removal by the secondary system.

The High SG Level Trip requires the following sensors and processors to be OPERABLE in MODE 1 and in MODE 2:

- SG Level (Narrow Range) sensors
- APUs, and
- ALUs.

This reactor trip ensures core integrity during transients involving a MFW malfunction that results in an increase in feedwater flow or in case of a SG level increase. The Analytical Limit~~LTSP~~ is sufficiently ~~below~~ above the full load operating value so as not to interfere with normal plant operation, but still ~~high~~ low enough to provide the required protection in the event of an abnormal condition.

The P13 permissive automatically enables the High SG Level Trip when the hot leg temperature is greater than or equal to 200ºF.

19. High Containment Pressure

In case of a postulated initiating event leading to water or steam discharge into the containment, a reactor trip is performed in order to ensure containment integrity and to adapt the reactor power to the capacity of the safety systems. This trip protects against the following postulated accidents or AOOs:

- Decrease in heat removal by the secondary system, and
- Decrease in reactor coolant inventory.

The High SG Pressure Trip requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2:

- Containment Equipment Compartment and Containment Service Compartment pressure sensors,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

In case of a postulated initiating event leading to water or steam discharge into the containment, a reactor trip is performed in order to ensure containment integrity and to adapt the reactor power to the capacity of the safety systems. The Analytical Limit~~LTSP~~ is ~~set~~ high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) and is not indicative of an abnormal condition. It is set low enough to initiate a reactor trip when an abnormal condition is indicated.

There are no permissives associated with this trip.

~~20. Manual Reactor Trip~~

~~A manual reactor trip signal can be generated from the SICS panel and the RSS. The manual trip signal from the RSS actuates a reactor trip through energizing the shunt coils of the main reactor trip breakers.~~

B. ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) FUNCTIONS

Each of the analyzed accidents or AOOs can be detected by one or more ESF Functions. One of the ESF Functions is the primary actuation signal for that accident. An ESF Function may be the primary actuation signal for more than one type of accident. An ESF Function may also be the secondary, or backup, actuation signal for one or more other accidents. The ESF protective functions are described below.

1. Turbine Trip on Reactor Trip

A turbine trip is required following any reactor trip in order to avoid a mismatch between primary and secondary power, which would result in excessive RCS cooldown with a potential return to critical conditions and power excursion.

The automatic Turbine Trip on Reactor Trip requires four divisions of the following sensors and processors to be OPERABLE in MODE 1:

- RTCB Position Indication sensor,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

A turbine trip is required following any reactor trip in order to avoid a mismatch between primary and secondary power. Such a mismatch could result in an RCS cooldown transient, with a potential inadvertent return to critical conditions. The one second time delay is an Analytical Limit.

There are no automatic permissives associated with this function.

2. Main Feedwater

a. MFW Full Load Closure on Reactor Trip (All SGs)

After a reactor trip check-back, a MFW full load isolation is required. This avoids a mismatch between primary and secondary power. Such a mismatch could result in an RCS cooldown transient, with a potential inadvertent return to critical conditions.

The automatic MFW Full Load Closure on Reactor Trip function requires four divisions of the following processors to be OPERABLE in MODE 1 and MODE 2 except when the MFW full load isolation valves are closed:

- RTCB Position Indication sensor,
- APUs, and
- ALUs.

There are no automatic permissives associated with this function.

b. MFW Full Load Closure on High SG Level (Affected SG)

In the case of an increasing SG level event, the MFW supply to the affected SG is isolated in order to avoid filling the SG, and subsequently introducing water into Main Steam line and MSRT.

This function mitigates an increase in heat removal from the secondary system.

The automatic MFW Full Load Closure on High SG Level function requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 and MODES 2 and 3, except when all MFW full load and low load isolation valves are closed:

- SG Level sensors,
- APUs, and
- ALUs.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The MFW Full Load Closure on High SG Level Analytical Limit~~LTSP~~ is ~~set~~ high enough to avoid spurious actuation but low enough in order to prevent water level in the SG from rising and entering the steam line.

The P13 permissive automatically enables the MFW Full Load Closure on High SG Level function when the hot leg temperature is greater than or equal to 200 ºF.

c. Startup and Shutdown Feedwater Isolation on SG Pressure Drop (All SGs)

The affected SG depressurizes for the listed events, a reactor trip is initiated on a SG pressure drop signal. Also, the Startup and Shutdown Feedwater (SSS) isolation and control valves close in all the SGs.

A complete Feedwater system isolation in the affected SG limits the coolant provided into the affected SG by the MFW/SSS. This action minimizes the mass and energy released into the containment and RCS cooldown.

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Steam system piping failure, and
- Feedwater system piping failure.

The automatic SSS Feedwater Isolation on SG Pressure Drop function requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 and MODES 2 and 3, except when all MFW full load and low load isolation valves are closed:

- SG pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is high enough to preclude spurious operation but low enough to terminate feedwater flow before overcooling of the primary system or depletion of secondary inventory.

There are no automatic permissives associated with this function.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

d. SSS Isolation on Low SG Pressure (All SGs)

The affected SG depressurizes in the event of a steam line or Feedwater pipe failure. In the event of a small secondary side break for which the SG pressure drop signal is never reached, this function also isolates the SSS supply to the affected SG. This action minimizes the mass and energy released into the containment.

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Steam system piping failure, and
- Feedwater system piping failure.

The automatic SSS Feedwater Isolation on Low SG Pressure function is required to be OPERABLE in:

– MODES 1,
– MODE 2, except when all MFW low load isolation valves are closed, and
– MODE 3 when the pressurizer pressure is greater than or equal to 2005 psia, except when all MFW low load isolation valves are closed.

The automatic SSS Feedwater Isolation on Low SG Pressure function requires four divisions of the following sensors and processors to be OPERABLE:

– SG pressure sensors,
– APUs, and
– ALUs.

The Analytical Limit~~LTSP~~ is high enough to preclude spurious operation but low enough to terminate feedwater flow before overcooling of the primary system or depletion of secondary inventory.

The P12 permissive automatically enables the SSS Isolation on Low SG Pressure function when the pressurizer pressure is greater than 2005 psia.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

e. SSS Isolation on High SG Level for Period of Time (Affected SGs)

During an increase in SG level after a reactor trip, the SSS systems are isolated in the affected SG in order to avoid the SG filling up and thus carryover of water into Main Steam line and subsequent water discharge by MSRT. This function mitigates Increase in Feedwater flow.

The automatic SSS Isolation on High SG Level for Period of Time function requires four divisions of the following sensors and processors to be OPERABLE in MODE 1 and MODES 2 and 3, except when all MFW low load isolation valves are closed:

- RTCB Position Indication,
- SG Level (Narrow Range) sensors,
- APUs, and
- ALUs.

The SSS Isolation on High SG Level for Period of Time Analytical Limit~~LTSP~~ is ~~set~~ high enough to avoid spurious actuation but low enough in order to prevent water level in the SGs from rising and entering the steam lines.

The P13 permissive automatically enables the SSS Isolation on High SG Level for Period of Time function when the hot leg temperature is greater than 200 ºF.

3. Safety Injection System Actuation

a. Low Pressurizer Pressure

In the event of a decrease in RCS water inventory, the makeup is supplied by the Medium Head Safety Injection (MHSI) in the high pressure phase of the event and the Low Head Safety Injection (LHSI) in the low pressure phase. For a potential overcooling event, the reactivity insertion is limited by the boron injection via the MHSI. Even if the boron injection is not required, MHSI injection is needed to stabilize the RCS pressure.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Safety Injection System (SIS) Actuation function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- MSLB,
- Feedwater Line Break,
- Inadvertent opening of a pressurizer pilot operated safety valve,
- Small break LOCA,
- Steam system piping failure, and
- Large break LOCA.

The automatic SIS Actuation on Low Pressurizer Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and MODE 3 with the pressurizer pressure greater than or equal to 2005 psia:

- Three Pressurizer Pressure (Narrow Range) sensors,
- Three divisions of APUs, and
- Three divisions of ALUs.

The Analytical Limit~~LTSP~~ for this function is set below the full load operating value for RCS pressure so as not to interfere with normal plant operation. However, the Analytical Limit~~setting~~ is high enough to provide an SIS actuation during an RCS depressurization.

The P12 permissive automatically enables the SIS Actuation on Low Pressurizer Pressure function when the pressurizer pressure is greater than or equal to 2005 psia.

The capability for manual initiation of the SIS is provided to the operator in the MCR. This manual initiation starts the four trains of SI. Four manual initiation controls are provided, any two of which will start the four SIS trains.

b. Low Delta P_{sat}

This function ensures SIS actuation in the hot and cold shutdown conditions with LHSI / Residual Heat Removal (RHR) in operation and at least one RCP operating.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This function mitigates the following postulated accidents or AOOs:

- Small break LOCA,
- Large break LOCA,
- Spurious opening of one Main Steam relief or safety valve,
- Inadvertent opening of a pressurizer pilot operated safety valve,
- Excessive increase in secondary steam flow, and
- MSLB.

The automatic SIS Actuation on Low Delta P_{sa} function requires four divisions of the following sensors and processors:

- Hot Leg Pressure (Wide Range) sensors,
- Hot Leg Temperature (Wide Range) sensors,
- APUs, and
- ALUs.

These sensors and processors are required to be OPERABLE in MODE 3 when Trip/Actuation Function B.3.a, SIS Actuation on Low Pressurizer Pressure, is disabled.

This function ensures SIS actuation in the hot and cold shutdown conditions with LHSI/RHR in operation and at least one of the RCPs are operating.

The Analytical Limit~~LTSP~~ for the Low Delta P_{sat} function is ~~set~~ high enough to avoid spurious operation but low enough to maintain core coverage in the event of an RCS pipe break.

The P12 permissive automatically enables the SIS Actuation on Low Delta P_{sat} function when the pressurizer pressure is less than or equal to 2005 psia. The P15 permissive automatically enables the SIS Actuation on Low Delta P_{sat} function when at least two RCPs are running, the hot leg pressure is greater than or equal to 464 psia, and when the hot leg temperature is greater than or equal to 356ºF.

The capability for manual initiation of the SIS is provided to the operator in the MCR. This manual initiation starts the four trains of SI. Four manual initiation controls are provided, any two of which will start the four SIS trains.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4. RCP Trip on Low Delta-Pressure across the RCP with SIS Actuation

In case of LOCA in combination with a SIS actuation, the RCPs are tripped to prevent their operation in scenarios where timing of the pump trip is related to maintaining core cooling.

This function mitigates the following postulated accidents or AOOs:

- Inadvertent opening of a PSRV, and
- Small break LOCA.

The automatic RCP Trip on Low Delta-Pressure across RCP with SIS Actuation function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- RCP Delta-Pressure sensors,
- RCP Current sensors,
- APUs, and
- ALUs.

The Analytical Limit ~~LTSP~~ for the RCP Trip on Low Delta-Pressure across RCP with SIS Actuation function is ~~set~~ high enough to avoid spurious operation but low enough to ensure core cooling is maintained.

There are no automatic permissives associated with this function.

5. Partial Cooldown on SIS Actuation

The partial cooldown consists of lowering the MSRT setpoint down to allow depressurization of the RCS by heat removal of the SGs. This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- MSLB,
- Inadvertent opening of a Pressurizer pilot operated safety valve, and
- Small break LOCA.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The automatic Partial Cooldown on SIS Actuation function requires four divisions of the following processors to be OPERABLE in MODES 1, 2, and 3:

- APUs, and
- ALUs.

~~The LTSP for the Partial Cooldown Actuation on SIS Actuation function is set high enough to avoid spurious operation but low enough to ensure adequate flow from the MHSI pumps to maintain core cooling.~~

The P14 permissive automatically enables the Partial Cooldown on SIS Actuation function when the hot leg pressure is greater than or equal to 464 psia and the hot leg temperature is greater than or equal to 356 ºF.

6. Emergency Feedwater System

a. Actuation on Low-Low SG Level (All SGs)

In case of loss of MFW, the Emergency Feedwater System (EFWS) is actuated to remove residual heat via secondary side. With an EFWS actuation signal, SG blowdown is also isolated to conserve SG inventory. This function mitigates the following postulated accidents or AOOs:

- Loss of normal feedwater flow,
- Feedwater system piping failure, and
- LOOP.

The automatic EFWS Actuation on Low-Low SG Level function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2 and 3:

- SG level (Wide Range) sensors,
- APUs, and
- ALUs.

This function ensures heat is removed from the primary system through the SGs in the event of a loss of MFW or feedwater line break, as indicated by low SG level. The Analytical Limit ~~LTSP~~ is high enough to provide an operating envelope that prevents unnecessary actuations but low enough to ensure sufficient make-up is provided to the SGs.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The P13 permissive automatically enables the EFWS Actuation on Low-Low SG Level function when the hot leg temperature is greater than or equal to 200ºF.

b. Actuation on LOOP and SIS Actuation (All SGs)

The LOOP results in a trip of the turbine, RCPs, and MFW pumps. The MFW and SSS supply cut off leads to a decrease in secondary side heat removal and the primary flow coast down further reduces the capacity of the primary coolant to remove heat from the core. As a result, primary and secondary pressures and temperatures increase. The heat is removed via MSRT and EFWS. With an EFWS actuation signal, SG blowdown is also isolated to conserve SG inventory.

This function mitigates the consequences of a Small Break LOCA.

The automatic EFWS Actuation on LOOP and SIS function requires four divisions of the following processors to be OPERABLE in MODES 1 and 2:

- 6.9 kV Bus Voltage sensors,
- APUs, and
- ALUs.

This function ensures heat is removed from the primary system through the SGs in the event of a LOCA concurrent with a LOOP.

There are no automatic permissives associated with this function.

c. Isolation on High SG Level (Affected SG)

In the case of an increasing SG level event, the EFWS supply to the affected SG is isolated in order to avoid filling the SG, and subsequently introducing water into Main Steam line and MSRT. This function precludes overfilling of the SG.

The automatic EFWS Isolation on High SG Level function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2 and 3:

- SG level (Wide Range) sensors,
- APUs, and
- ALUs.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This function ensures the SGs are not overfilled, which could allow radioactive water to be discharged through the MSRTs. The ~~LTSP~~ Design Limit is high enough to provide an operating envelope that prevents unnecessary isolations but low enough to ensure the SGs are not over-filled.

The P13 permissive automatically enables the EFWS Isolation on High SG Level function when the hot leg temperature is greater than or equal to 200 ºF.

7. Main Steam Relief Train

a. Actuation on High SG Pressure

In the event of a loss of the secondary side heat sink, the residual heat is removed through the steam relief valves to the atmosphere. This is done by the MSRT. The MSRT also ensures SG overpressure protection, minimizes the actuation of the Main Steam Safety Valves (MSSVs), which reduces the risk of a stuck open safety relief valve.

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Total loss of load and/or turbine trip
- Loss of main heat sink (condenser),
- Inadvertent closure of a Main Steam Isolation Valve (MSIV),
- MSLB,
- RCP seizure (locked rotor) or RCP shaft break., and
- Feedwater system piping failure.

The automatic MSRT Actuation on High SG Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- SG Pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ for the MSRT Actuation on High SG Pressure function is set high enough to avoid spurious operation and low enough to open and relieve SG pressure before over pressurization limits are reached.

There are no automatic permissives associated with this function.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Isolation on Low SG Pressure

The Main Steam Relief Isolation Valves (MSRIVs) are opened during events in order to control pressure in the SGs. In order to prevent a stuck open Main Steam Relief Control Valve from causing an RCS cooldown and a risk of return to critical conditions, the MSRT is isolated.

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Loss of main heat sink (condenser),
- Inadvertent Opening of SG Safety or Relief Valve, and
- MSLB.

The automatic MSRT Isolation on Low SG Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and MODE 3 with the pressurizer pressure is greater than or equal to 2005 psia:

- SG pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ for the MSRT Isolation on Low SG Pressure function is set low enough to avoid spurious operation and high enough to limit the rate of RCS cooldown.

The P12 permissive automatically enables the MSRT Isolation on Low SG Pressure function when the pressure is greater than or equal to 2005 psia.

8. MSIV Closure

a. Closure on SG Pressure Drop (All SGs)

In case of a secondary side Steam Line or Feedwater system pipe break, the affected SG depressurizes. This function isolates all four SGs in order to:

- Prevent draining of unaffected SG,
- Limit return to criticality conditions due to a overcooling transient,
- Limit the release of radioactivity, and
- Limit mass and energy releases into the containment.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Spurious opening of one SG safety or relief valve,
- Steam system piping failure, and
- Feedwater system piping failure.

The automatic MSIV Closure on SG Pressure Drop function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- SG Pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ for the MSIV Closure on SG Pressure Drop function is ~~set~~ high enough to avoid SG pressure fluctuations during normal operation and low enough to isolate a SG and limit the blowdown to the value assumed in the safety analysis.

There are no automatic permissives associated with this function.

b. Closure on Low SG Pressure (All SGs)

For most Main Steam Line or Feedwater pipe breaks, the affected SG depressurizes. For small breaks, the setpoint for MSIV closure on SG pressure drop may not be reached. This function isolates all four SG on the main steam side in the event of a secondary side break in order to:

- Prevent draining of unaffected SGs,
- Limit the return to critical conditions due to a overcooling transient,
- Limit the release of radioactivity, and
- Limit mass and energy releases into the containment.

This function mitigates the following postulated accidents or AOOs:

- Excessive increase in secondary steam flow,
- Spurious opening of one SG safety or relief valve,
- Steam system piping failure, and
- Feedwater system piping failure.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The automatic MSIV Closure on Low SG Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1 and 2 and MODE 3, except when all MSIVs are closed:

- SG pressure sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ for the MSIV Closure on Low SG Pressure function is ~~set~~ high enough to avoid SG pressure fluctuations during normal operation and low enough to isolate a SG and limit the blowdown to the value assumed in the safety analysis.

The P12 permissive automatically enables the MSIV Closure on Low SG Pressure function when the pressurizer pressure is greater than or equal to 2005 psia.

9. Containment Isolation

a. Isolation (Stage 1) on High Containment Pressure

In case of a LOCA, the containment has to be isolated in order to prevent release of radioactivity to the environment. Safeguards Building HVAC is also reconfigured to process air through High Efficiency Particulate Air (HEPA) filters to ensure 10 CFR 50.34 and 10 CFR 100.21 limits are not exceeded.

The automatic Stage 1 Containment Isolation on High Containment Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- Containment Service Compartment Pressure monitors,
- Containment Equipment Compartment Pressure monitors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ for the Stage 1 Containment Isolation on High Containment Pressure function is ~~set~~ high enough to avoid spurious operation but low enough to ensure offsite dose consequences are maintained below 10 CFR 50.34 and 10 CFR 100.21 limits.

There are no automatic permissives associated with this function.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Isolation (Stage 1) on SIS Actuation

In case of the listed events, the containment has to be isolated in order to prevent release of radioactivity to the environment. Safeguards Building HVAC is also reconfigured to process air through HEPA filters to ensure 10 CFR 50.34 and 10 CFR 100.21 limits are not exceeded.

This function mitigates the following postulated accidents or AOOs:

- Inadvertent opening of a pressurizer pilot operated safety valve, and
- LOCA.

The automatic Stage 1 Containment Isolation on SIS Actuation function requires four divisions of the following processors to be OPERABLE in MODES 1, 2, 3, and 4:

- APUs, and
- ALUs.

There are no automatic permissives associated with this function.

c. Isolation (Stage 2) on High-High Containment Pressure

In case of a LOCA, the containment has to be isolated in order to prevent release of radioactivity to the environment.

This function mitigates the following postulated accidents or AOOs:

- Inadvertent opening of a pressurizer pilot operated safety valve, and
- LOCA.

The automatic Stage 2 Containment Isolation on High-High Containment Pressure function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- Containment Service Compartment Pressure monitors,
- Containment Equipment Compartment Pressure monitors,
- APUs, and
- ALUs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The ~~LTSP~~ Design Limit for the Stage 2 Containment Isolation on High-High Containment Pressure function is ~~set~~ high enough to avoid spurious operation but low enough to ensure offsite dose consequences are maintained below 10 CFR 50.34 and 10 CFR 100.21 limits.

There are no automatic permissives associated with this function.

d. Isolation (Stage 1) on High Containment Radiation

In case of a significant release of radioactivity into the containment, the containment is isolated to ensure 10 CFR 50.34 and 10 CFR 100.21 limits are not exceeded.

This function mitigates the following postulated accidents or AOOs:

- Rod ejections,
- LOCA,
- MSLB inside containment, and
- Feedwater line break inside containment.

The automatic Stage 1 Containment Isolation on High Containment Radiation function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, 3, and 4:

- Containment High Range radiation monitors,
- APUs, and
- ALUs.

The ~~LTSP~~ Design Limit for the Stage 1 Containment Isolation on High Containment Radiation function is ~~set~~ high enough to avoid spurious operation but low enough to ensure offsite dose consequences are maintained below 10 CFR 50.34 and 10 CFR 100.21 limits.

There are no automatic permissives associated with this function.

10. Emergency Diesel Generator

a. Start on Degraded Grid Voltage

Following the detection of degraded voltage for a period of time on one 6.9 kV bus, the EDG associated with that bus is automatically started. This function mitigates a LOOP, which is assumed to occur independently or concurrently with postulated accidents and AOOs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The automatic EDG Start on Degraded Grid Voltage requires four divisions of the following processors to be OPERABLE in MODES 1, 2, 3, and 4 or when the associated EDG is required to be OPERABLE in accordance with LCO 3.8.2, “AC Sources - Shutdown”:

- 6.9 kV voltage sensors,
- APUs, and
- ALUs.

This function ensures AC Power is available to mitigate a postulated concurrent design basis event.

The ~~LTSP~~ Design Limit for the EDG Start on Degraded Grid Voltage is ~~set~~ high enough to avoid spurious operation but low enough to ensure that power is provided to ESF functions in the time-frame assumed in the accident analyses.

There are no automatic permissives associated with this function.

b. Start on LOOP

Following a LOOP on one 6.9 kV bus, the EDG associated with that bus is automatically started. This function mitigates a LOOP, which is assumed to occur independently or concurrently with postulated accidents and AOOs.

The automatic EDG Start on LOOP requires four divisions of the following processors to be OPERABLE in MODES 1, 2, 3, and 4 or when the associated EDG is required to be OPERABLE in accordance with LCO 3.8.2, “AC Sources - Shutdown”:

- 6.9 kV voltage sensors,
- APUs, and
- ALUs.

This function ensures AC Power is available to mitigate a postulated concurrent design basis event.

The ~~LTSP~~ Design Limit for the EDG Start on LOOP is ~~set~~ high enough to avoid spurious operation but low enough to ensure that power is provided to ESF functions in the time-frame assumed in the accident analyses.

There are no automatic permissives associated with this function.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

11. Chemical and Volume Control System Charging Line Isolation

a. Isolation on High-High Pressurizer Level

The isolation of the CVCS Charging Line on High-High Pressurizer Level is required to avoid filling of the pressurizer and subsequent water overflow through the safety valves.

This function protects against a CVCS malfunction that causes an increase in RCS water inventory.

The automatic CVCS Charging Line Isolation on High-High Pressurizer Level function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, and 3:

- Pressurizer Level (Narrow Range) sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~ is low enough to initiate appropriate mitigative actions in time to prevent the pressurizer from overfilling during the CVCS Malfunction event that may increases RCS inventory, but high enough to prevent spurious operations.

The P17 permissive automatically disables the CVCS Charging Line Isolation on High-High Pressurizer Level function when the cold leg temperature is less than or equal to 248 ºF.

b. Isolation on ADM - Shutdown Condition (RCP not operating)

The ADM function in the Shutdown Condition mitigates a dilution event where no RCPs are in operation. This function ensures that:

- The dilution is stopped when the protection is actuated, and
- The core remains sub-critical.

The automatic CVCS Charging Line Isolation on ADM - Shutdown Condition (RCP not operating) function is required to be OPERABLE in:

- MODES 5, with two or less RCPs in operation, and
- MODES 6.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The automatic CVCS Charging Line Isolation on ADM - Shutdown Condition (RCP not operating) function requires the following sensors and processors:

- Boron Concentration - CVCS Charging Line sensors (4 divisions),
- Boron Temperature - CVCS Charging Line sensors (4 divisions),
- APUs (4 divisions), and
- ALUs (Divisions 1 and 4).

The Analytical Limit ~~LTSP~~ is high enough to provide an operating envelope that prevents unnecessary isolations but low enough to mitigate a dilution event in the shutdown condition where the RCPs are not in operation.

This function is required to be accompanied by Permissive P7, which represents a RCP speed shutdown condition, or an ATWS signal.

c. Isolation on ADM - Standard Shutdown Conditions

This function mitigates a homogeneous dilution event in the standard shutdown states where the RCPs are in operation. This function ensures that:

- The dilution is stopped when the protection is actuated, and
- The core remains sub-critical.

The automatic CVCS Charging Line Isolation on ADM - Standard Shutdown Conditions function is required to be OPERABLE in:

- MODES 3, with thee or more RCPs in operation,
- MODES 4, with thee or more RCPs in operation, and
- MODES 5, with thee or more RCPs in operation.

The automatic CVCS Charging Line Isolation on ADM - Standard Shutdown Conditions function requires the following sensors and processors:

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

- Boron Concentration - CVCS Charging Line sensors (4 divisions),
- Boron Temperature - CVCS Charging Line sensors (4 divisions),
- CVCS Charging Line Flow sensors (4 divisions),
- Cold Leg Temperature (Wide Range) sensors (4 divisions),
- APUs (4 divisions), and
- ALUs (Divisions 1 and 4).

The Analytical Limit~~LTSP~~ is high enough to provide an operating envelope that prevents unnecessary isolations but low enough to mitigate a dilution event in the shutdown condition where the RCPs are in operation.

This function is required to be accompanied by a permissive signal, P8, which represents a reactor shutdown condition as indicated by RCCA position indication and disabled by the Permissive P7, which represents a RCP shutdown condition.

12.a and 12.b. PSRV Actuation - First and Second Valve

The integrity of the reactor pressure vessel must be ensured under all plant conditions. At low coolant temperature, the cylindrical part of the vessel could fail by brittle fracture before the design pressure of the RCS is reached. Therefore the low-temperature overpressure protection (LTOP) is ensured by opening of the PSRVs.

This function mitigates a low temperature overpressure event.

The automatic PSRVs Actuation function requires four divisions of the following processors to be OPERABLE when the PSRVs are required to be OPERABLE by LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)"

- Hot Leg Pressure (Wide Range) sensors,
- APUs, and
- ALUs.

The Analytical Limit~~LTSP~~s for the PSRV Actuation function are high enough to prevent spurious operation but low enough to prevent RCS over pressurization.

The P17 permissive automatically enables the PSRV Actuation function when the cold leg temperature is less than or equal to 248º F.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

13. Control Room HVAC Reconfiguration to Recirculation Mode on High Intake Activity

In case of a significant release of radioactivity, the Control Room HVAC is reconfigured to ensure 10 CFR 50.34 limits are not exceeded.

This function mitigates the following postulated accidents or AOOs:

- Rod ejections,
- LOCA, and
- Line breaks outside containment.

The automatic Control Room HVAC Reconfiguration to Recirculation Mode on High Intake Activity function requires four divisions of the following sensors and processors to be OPERABLE in MODES 1, 2, 3, 4, 5, 6, and during the movement of irradiated fuel assemblies:

- Control Room HVAC Intake Activity radiation monitors,
- APUs, and
- ALUs.

The ~~LTSP~~ Design Limit for the Control Room HVAC Reconfiguration to Recirculation Mode on High Intake Activity function is ~~set~~ high enough to avoid spurious operation but low enough to ensure offsite dose consequences are maintained below 10 CFR 50.34 limits.

There are no automatic permissives associated with this function.

C. PROTECTION SYSTEM PERMISSIVES

Protection System permissives are provided to ensure reactor trips and ESF are in the correct configuration for the current unit status. They back up operator actions to ensure Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the permissive Functions do not need to be OPERABLE when the associated reactor trip or ESF functions are outside the applicable MODES. The automatic permissives are:

1. P2 - Flux (Power Range) Measurement Higher than First Threshold

The P2 permissive is representative of PRD neutron flux measurements higher than a low-power setpoint value. The P2 setpoint value corresponds to the value below which transients do not lead to risk of DNB (10% RTP).

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The P2 permissive is utilized in the following reactor trips or ESF functions:

- Reactor Trip 1.a - Low DNBR,
- Reactor Trip 1.b - Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c - Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d - Low DNBR - High Quality,
- Reactor Trip 1.e - Low DNBR - High Quality and Imbalance or Rod Drop,
- Reactor Trip 2 - High Linear Power Density,
- Reactor Trip 6.b - Low RCS Flow Rate in Two Loops,
- Reactor Trip 7 - Low RCP Speed, and
- Reactor Trip 10 - Low Pressurizer Pressure.

To generate the permissive, neutron flux measurements from the PRDs are compared to the setpoint. When two out of four measurements are greater than the setpoint, the permissive is validated. Otherwise, it is inhibited.

The value of the permissive was selected such that AOOs do not challenge the DNBR or centerline melt limits when they occur at a core power level below the permissive value.

2. P3 - Flux (Power Range) Measurement Higher than Second Threshold

The P3 permissive is representative of PRD neutron flux measurements higher than an intermediate power setpoint value. The P3 setpoint value corresponds to the value below which loss of one reactor coolant pump does not lead to risk of DNB (70% Nuclear Power).

The P3 permissive is utilized in Reactor Trip 6.a - Low-Low RCS Flow Rate in One Loop.

To generate the permissive, neutron flux measurements from the PRDs are compared to the setpoint. When two out of four measurements are greater than the setpoint, the permissive is validated.

The value of the permissive was selected such that AOOs and postulated accidents that consider a loss of one RCP do not challenge the DNBR limit when they occur at a core power level below the permissive value (70% RTP).

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. P5 - Flux (Intermediate Range) Measurement Higher than Threshold

The P5 permissive is representative of intermediate range detector (IRD) neutron flux measurements above a low-power setpoint value. The P5 setpoint value corresponds to the boundary between the operating ranges of the source range detectors and intermediate range detectors (greater than or equal to 10^{-5}% power as shown on the IRDs).

The P5 permissive is utilized in the following reactor trips or ESF functions:

- Reactor Trip 4 - High Core Power Level, and
- Reactor Trip 5 - Low Saturation Margin.

To generate the permissive, neutron flux measurements from the IRDs are compared to the setpoint. When two out of four of the measurements are greater than the setpoint, the permissive is validated.

The value of the permissive defines the boundary between the operating range of the source range detectors and the operating range of the intermediate range detectors.

4. P6 - Thermal Core Power Higher than Threshold

The P6 permissive is representative of core thermal power above a low-power setpoint value corresponding to the boundary between the operating ranges of the IRDs and the PRDs (10% RTP).

The P6 permissive is utilized in the following reactor trips or ESF functions:

- Reactor Trip 8 - High Neutron Flux (Intermediate Range), and
- Reactor Trip 9 - Low Doubling Time (Intermediate Range).

Hot leg pressure measurements, hot leg temperature measurements, and cold leg temperature measurements are used to calculate core thermal power. These calculated core thermal power levels are compared to the setpoint. When three out of four of the calculated core thermal power levels are greater than the setpoint, the operator is prompted to manually validate the permissive.

The value of the permissive was selected at the boundary between the operating range of the intermediate range detectors and the power range detectors.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5. P7 - RCP Speed Lower than Threshold

The P7 permissive defines when reactor coolant pumps (RCPs) are no longer in operation. The P7 permissive is utilized in the following reactor trips or ESF functions:

- ESF 11.b - CVCS Charging Line Isolation on ADM at Shutdown Condition (RCP not operating), and
- ESF 11.c - CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions.

The RCP speed measurements (one per RCP) are compared to a setpoint (91% nominal speed). When two out of four of the measurements are less than the setpoint, the permissive is validated (i.e., indicates that two or more RCPs are turned off).

The value of the permissive was selected to establish the requirements for anti-dilution mitigation in a timely manner.

6. P8 - Shutdown RCCA Position Lower than Threshold

The P8 permissive defines the shutdown state with all rods in (ARI). The P8 permissive is utilized in ESF 11.c - CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions.

RCCA Bottom Position Indicator sensors are acquired in four different electrical divisions. For each division, when all rods in the shutdown banks reach the lower end position, a signal is generated. When two out of four of divisions indicate all rods in, the permissive is validated.

The P8 Permissive is characteristic of a shutdown state with ARI. With an ARI condition, this permissive enables the Anti-dilution in Standard Shutdown States function and inhibits the Anti-dilution in Power Condition” function.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7. P12 - Pressurizer Pressure Lower than Threshold

The P12 permissive defines the transition from hot shutdown to cold shutdown with respect to RCS pressure. The P12 permissive is utilized in the following reactor trips or ESF functions:

- Reactor Trip 12 - High Pressurizer Level,
- Reactor Trip 13 - Low Hot Leg Pressure,
- Reactor Trip 15 - Low SG Pressure Trip,
- ESF 2.d - SSS Isolation on Low SG Pressure (All SGs),
- ESF 3.a - SIS Actuation on Low Pressurizer Pressure,
- ESF 3.b - SIS Actuation on Low Delta Psat,
- ESF 7.b - MSRT Isolation on Low SG Pressure
- ESF 8.b - MSIV Closure on Low SG Pressure (All SGs), and
- ESF 9.b - Containment Isolation (Stage 1) on SIS Actuation.

Pressurizer pressure measurements are compared to the P12 setpoint (2005 psia). The low SG pressure and low hot leg pressure reactor trip functions are automatically activated when the pressurizer pressure rises above the P12 permissive value.

The Permissive P12 reflects the transition from hot shutdown to cold shutdown. P12 ensures cooling by Main Steam Bypass or MSRT down to the LHSI/RHR connection temperature and to be able to depressurize the reactor coolant system to LHSI/RHR connection pressure without actuation of SIS.

8. P13 - Hot Leg Temperature Lower than Threshold

The P13 permissive defines when steam generator draining and filling operations are allowed. The P13 permissive is utilized in the following reactor trips or ESF functions:

- Reactor Trip17 - Low SG Level,
- Reactor Trip 18 - High SG Level,
- ESF 2.b - MFW Full Load Closure on High SG Level (Affected SGs)
- ESF 2.e - SSS Isolation on High SG Level for Period of Time (Affected SGs),
- ESF 6.a - EFWS Actuation on Low-Low SG Level (All SGs), and
- ESF 6.c - EFWS Isolation on High SG Level (Affected SG).

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Hot leg temperature (WR) measurements are compared to the P13 setpoint (200ºF).

The value of the permissive was selected in order to permit draining and filling operations during shutdown and LHSI/RHR in operation without generating protection signals.

9. <u>P14 - Hot Leg Pressure and Hot Leg Temperature Lower than Thresholds</u>

The P14 permissive defines when the residual heat removal system is allowed to be connected to the RCS. The P14 permissive is utilized in ESF 5 - Partial Cooldown Actuation on SIS Actuation.

At pressures and temperatures below the setting of the P14 permissive (464 psia and 356 ºF), operation of the LHSI/RHR system is allowed.

This permissive is manually controlled.

10. <u>P15 - RCPs Shutdown and P14</u>

The P15 permissive defines when SI actuation due to delta Psat is disabled and SI actuation due to low loop level is enabled.

The P15 permissive is utilized in the following reactor trips or ESF functions:

- ESF 3.b - SIS Actuation on Low Delta Psat, and
- ESF 9.b - Containment Isolation (Stage 1) on SIS Actuation.

The value for Permissive P15 (50% no load current and P14 is true) represents the threshold for switching from the SIS Actuation on Low Delta P_{sat} protection to protection via the SIS Actuation on Low RCS Loop Level.

11. <u>P17 - Cold Leg Temperature Lower than Threshold</u>

The P17 permissive corresponds to the temperature conditions where brittle fracture protection is required. The P17 permissive is utilized in the following reactor trips or ESF functions:

- ESF 12.a - PSRV Actuation - First Valve, and
- ESF 12.b - PSRV Actuation - Second Valve.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The value for Permissive P17 is the threshold for activation of cold overpressure mitigation systems.

D. SENSORS, MANUAL ACTUATION SWITCHES, SIGNAL PROCESSORS, AND ACTUATION DEVICES

The relationship between sensors, manual actuation switches, signal processors, and actuation devices is provided below:

SENSORS

1. 6.9 kV Bus Voltage

Three 6.9 kV Bus Voltage sensors per EDG are required to be OPERABLE in MODES 1, 2, 3, 4, and when the associated EDG is required to be OPERABLE by LCO 3.8.2. These sensors support the following functions:

- ESF 6.b: EFWS Actuation on LOOP and SIS Actuation (All SGs),
- ESF 10.a: EDG Start on Degraded Grid Voltage, and
- ESF 10.b: EDG Start on LOOP.

2. Boron Concentration - CVCS Charging Line

Four Boron Concentration - CVCS Charging Line sensors are required to be OPERABLE in MODES 3 and 4 with three or more RCPs in operation and in MODES 5 and 6. These sensors support the following functions:

- ESF 11.b: CVCS Charging Line Isolation on ADM at Shutdown Condition (RCP not operating), and
- ESF 11.c: CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Boron Temperature - CVCS Charging Line

Four Boron Temperature - CVCS Charging Line sensors are required to be OPERABLE in MODES 3 and 4 with three or more RCPs in operation and in MODES 5 and 6. These sensors support the following functions:

- ESF 11.b: CVCS Charging Line Isolation on ADM at Shutdown Condition (RCP not operating), and
- ESF 11.c: CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions.

4. CVCS Charging Line Flow

Four CVCS Charging Line Flow sensors are required to be OPERABLE in MODES 3, 4, and 5 when three or more RCPs are in operation. These sensors support ESF 11.c: CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions.

5. Cold Leg Temperature (Narrow Range)

Four Cold Leg Temperature (Narrow Range) sensors are required to be OPERABLE when RTP is greater than or equal to 10%. These sensors support the following functions and Permissives:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop, and
- Permissive P6: Thermal Core Power Higher than Threshold.

6. Cold Leg Temperature (Wide Range)

Four Cold Leg Temperature (Wide Range) sensors are required to be OPERABLE in:

- MODE 1,
- MODE 2, when power is greater than or equal to 10^{-5}% as shown on the intermediate range detectors, and in
- MODES 3, 4, 5, and 6 with three or more RCPs in operation.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

These sensors support the following functions and Permissives:

- Reactor Trip 4: High Core Power Level,
- Reactor Trip 5: Low Saturation Margin,
- ESF 11.c: CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions, and
- Permissive P17: Cold Leg Temperature Lower than Threshold.

7. Containment Pressure

Four Containment Equipment Compartment Containment and Service Compartment Pressure sensors per area are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions:

- Reactor Trip 19: High Containment Pressure,
- ESF 9.a: Containment Isolation (Stage 1) on High Containment Pressure, and
- ESF 9.c: Containment Isolation (Stage 2) on High-High Containment Pressure.

8. Hot Leg Pressure (Wide Range)

Four Hot Leg Pressure (Wide Range) sensors are required to be OPERABLE in Modes 1, 2, and 3, and when the PSRVs are required to be OPERABLE per LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)." These sensors support the following functions and Permissives:

- Reactor Trip 4: High Core Power Level,
- Reactor Trip 5: Low Saturation Margin,
- Reactor Trip 13: Low Hot Leg Pressure,
- ESF 3.b: SIS Actuation on Low Delta P_{sat},
- ESF 12.a: PSRV Actuation - First Valve,
- ESF 12.b: PSRV Actuation - Second Valve,
- Permissive P6: Thermal Core Power Higher than Threshold,
- Permissive P14: Hot Leg Pressure and Hot Leg Temperature Lower than Thresholds, and
- Permissive P15: RCPs Shutdown and P14.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

9. Hot Leg Temperature (Narrow Range)

Four Hot Leg Temperature (Narrow Range) sensors in each of four divisions (16 total) are required to be OPERABLE in MODE 1 and MODE 2 when power is greater than or equal to 10^{-5}% as shown on the intermediate range detectors. These sensors support the following functions and Permissives:

- Reactor Trip 4: High Core Power Level,
- Reactor Trip 5: Low Saturation Margin, and
- Permissive P6: Thermal Core Power Higher than Threshold.

10. Hot Leg Temperature (Wide Range)

Four Hot Leg Temperature (Wide Range) sensors are required to be OPERABLE in MODE 3 when Trip/Actuation Function B.3.a, SIS Actuation on Low Pressurizer Pressure, is disabled. These sensors support the following functions and Permissives:

- ESF 3.b: SIS Actuation on Low Delta P_{sat},
- Permissive P13: Hot Leg Temperature Lower than Threshold,
- Permissive P14: Hot Leg Pressure and Hot Leg Temperature Lower than Thresholds, and

11. Intermediate Range

Four Intermediate Range sensors are required to be OPERABLE in:

- MODE 1, when power is less than or equal to 10% RTP,
- MODE 2, and in
- MODES 3 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted.

These sensors support the following functions and Permissives:

- Reactor Trip 8: High Neutron Flux (Intermediate Range),
- Reactor Trip 9: Low Doubling Time (Intermediate Range), and
- Permissive P5: Flux (Intermediate Range) Measurement Higher than Threshold.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

12. Power Range

Two Power Range sensors per division (8 total) are required to be OPERABLE in MODES 1 and 2, and in MODE 3 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted. These sensors support the following functions and Permissives:

- Reactor Trip 3: High Neutron Flux Rate of Change,
- Permissive P2: Flux (Power Range) Measurement Higher than First Threshold, and
- Permissive P3: Flux Measurement (Power Range) Higher than Second Threshold.

13. Pressurizer Level (Narrow Range)

Four Pressurizer Level (Narrow Range) sensors are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions:

- Reactor Trip 12: High Pressurizer Level, and
- ESF 11.a: CVCS Charging Line Isolation on High-High Pressurizer Level.

14. Pressurizer Pressure (Narrow Range)

Four Pressurizer Pressure (Narrow Range) sensors are required to be OPERABLE in MODES 1 and 2 and MODE 3 when the pressurizer pressure is less than or equal to 2005 psia. These sensors support the following functions and Permissives:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop,
- Reactor Trip 10: Low Pressurizer Pressure,
- Reactor Trip 11: High Pressurizer Pressure,
- ESF 3.a: SIS Actuation on Low Pressurizer Pressure, and
- Permissive P12: Pressurizer Pressure Lower than Threshold.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

15. Radiation Monitor - Containment High Range

Four Containment High Range Radiation Monitors are required to be OPERABLE in MODES 1, 2, 3, and 4. These sensors support ESF 9.d: Containment Isolation (Stage 1) on High Containment Radiation.

16. Radiation Monitor - Control Room HVAC Intake Activity

Four Control Room HVAC Intake Activity Radiation Monitors are required to be OPERABLE in MODES 1, 2, 3, 4, 5, 6, and during the movement of irradiated fuel assemblies. The monitors are not required to be OPERABLE when the associated train is in the recirculation mode. These sensors support ESF 13: Control Room HVAC Reconfiguration to Recirculation Mode on High Intake Activity.

17. RCP Current

Three RCP Current sensors per RCP (12 total) are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions and Permissives:

- ESF 4: RCP Trip on Low Delta P across RCP with SIS Actuation, and
- Permissive P15: Hot Leg Pressure and Hot Leg Temperature Lower than Thresholds and Reactor Coolant Pumps Shutdown.

18. RCP Delta P Sensors

Two RCP Delta-Pressure sensors per pump (8 total) are required to be OPERABLE in MODES 1, 2, and 3. These sensors support ESF 4: RCP Trip on Low Delta P across RCP with SIS Actuation.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

19. RCP Speed

Four RCP Speed sensors are required to be OPERABLE when RTP is greater than or equal to 10%. These sensors support the following functions and Permissives:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop,
- Reactor Trip 7: Low RCP Speed, and
- Permissive P7: RCP Speed Lower than Threshold.

20. RCS Loop Flow

Four RCS Loop Flow sensors per loop (16 total) are required to be OPERABLE in MODE 1 and in MODE 2 when power is greater than or equal to 10^{-5}% as shown on the intermediate range detectors. These sensors support the following functions and Permissives:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop,
- Reactor Trip 4: High Core Power Level,
- Reactor Trip 5: Low Saturation Margin,
- Reactor Trip 6a: Low-Low RCS Loop Flow Rate in One Loop,
- Reactor Trip 6b: Low RCS Loop Flow Rate in Two Loops, and
- Permissive P6: Thermal Core Power Higher than Threshold.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

21. RTCB Position Indication

Four RTCB Position Indication sensors are required to be OPERABLE in MODE 1 and in MODES 2 and 3 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted. These sensors support the following functions:

- ESF 1: Turbine Trip on Reactor Trip,
- ESF 2.a: MFW Full Load Closure on Reactor Trip (All SGs), and
- ESF 2.e: MFW and SSS Isolation on High SG Level for Period of Time (Affected SGs).

22. Self-Powered Neutron Detectors

Seventy two SPNDs are required to be OPERABLE when RTP is greater than or equal to 10%. These sensors support the following functions:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop, and
- Reactor Trip 2: High Linear Power Density.

23. SG Level (Narrow Range)

Four SG Level (Narrow Range) sensors per SG (16 total) are required to be OPERABLE in MODE 1 and in MODES 2 and 3, except when all MFW isolation valves are closed. These sensors support the following functions:

- Reactor Trip 17: Low SG Level,
- Reactor Trip 18: High SG Level,
- ESF 2.b: MFW Full Load Closure on High SG Level (Affected SGs), and
- ESF 2.e: SSS Isolation on High SG Level for Period of Time (Affected SGs).

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

24. SG Level (Wide Range)

Four SG Level (Wide Range) sensors per SG (16 total) are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions:

- ESF 6.a: EFWS Actuation on Low-Low SG Level (All SGs), and
- ESF 6.c: EFWS Isolation on High SG Level (Affected SG).

25. SG Pressure

Four SG Pressure sensors per SG (16 total) are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions:

- Reactor Trip 14: SG Pressure Drop,
- Reactor Trip 15: Low SG Pressure,
- Reactor Trip 16: High SG Pressure,
- ESF 2.c: SSS Isolation on SG Pressure Drop (All SGs),
- ESF 2.d: SSS Isolation on Low SG Pressure (All SGs),
- ESF 7.a: MSRT Actuation on High SG Pressure,
- ESF 7.b: MSRT Isolation on Low SG Pressure,
- ESF 8.a: MSIV Closure on SG Pressure Drop (All SGs), and
- ESF 8.b: MSIV Closure on Low SG Pressure (All SGs).

MANUAL ACTUATION SWITCHES

1. Reactor Trip

Four manual Reactor Trip switches are required to be OPERABLE in MODES 1 and 2 and in MODES 3, 4, and 5 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted. These sensors support all reactor trip functions.

2. SIS Actuation

Four manual SIS Actuation switches are required to be OPERABLE in MODES 1, 2, 3, and 4. These sensors support the following functions:

- ESF 3.a: SIS Actuation on Low Pressurizer Pressure,
- ESF 3.b: SIS Actuation on Low Delta P_{sat}.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. SG Isolation

Four manual SG Isolation switches per SG (16 total) are required to be OPERABLE in MODES 1, 2, and 3. These sensors support the following functions:

- ESF 2.b: MFW Full Load Closure on High SG Level (Affected SGs);
- ESF 2.c. SSS Isolation on SG Pressure Drop (All SGs);
- ESF 5: Partial Cooldown Actuation on SIS Actuation;
- ESF 6.a: EFWS Actuation on Low-Low SG Level (All SGs);
- ESF 6.c. EEFWS Isolation on High SG Level (Affected SG); and
- ESF 8.a: MSIV Closure on SG Pressure Drop (All SGs).

SIGNAL PROCESSORS

1. Remote Acquisition Units

Two RAUs per division (8 total) are required to be OPERABLE when RTP is greater than or equal to 10%. These signal processors support the following functions:

- Reactor Trip 1.a: Low DNBR,
- Reactor Trip 1.b: Low DNBR and Imbalance or Rod Drop,
- Reactor Trip 1.c: Variable Low DNBR and Rod Drop,
- Reactor Trip 1.d: Low DNBR - High Quality,
- Reactor Trip 1.e: Low DNBR - High Quality and Imbalance or Rod Drop, and
- Reactor Trip 2: High Linear Power Density.

2. Acquisition and Processing Units

Five APUs per division (20 total) are required to be OPERABLE in accordance with the supported functions as shown in Table 3.3.1-2. These signal processors support the reactor trip, ESF functions, and Permissives.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Actuation Logic Units

Four ALUs per division (16 total) are required to be OPERABLE in MODES 1, 2, 3, 4, 5, 6, and during the movement of irradiated fuel assemblies. These signal processors support the reactor trip, ESF functions and Permissives.

ACTUATION DEVICES

1. RCP Bus and Trip Breakers

Two RCP Bus and Trip Breakers per pump (8 total) are required to be OPERABLE in MODES 1, 2, 3, and 4. These actuation devices support ESF 4: RCP Trip on Low Delta P across RCP with SIS.

2. Reactor Trip Circuit Breakers

Four RTCBs are required to be OPERABLE in MODES 1 and 2 and in MODE 3 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted. These actuation devices support the reactor trip functions.

3. Reactor Trip Contactors

Twenty three sets of four Reactor Trip Contactors (92 total) are required to be OPERABLE in MODES 1 and 2 and in MODE 3 with the RCSL System capable of withdrawing a RCCA or one or more RCCAs not fully inserted. These actuation devices support the reactor trip functions.

ACTIONS

The most common causes of division inoperability are outright failure or drift of the sensor sufficient to exceed the tolerance allowed by the plant specific setpoint analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CALIBRATION when the sensor is set up for adjustment to bring it to within specification. ~~If the trip setpoint is non-conservative with respect to the Allowable Value, the division is declared inoperable immediately, and the appropriate Condition(s) must be entered immediately.~~ The SCP ensures that the divisions are performing as expected by confirming that the drift and other related errors are consistent with the supporting setpoint calculations.

ACTIONS (continued)

~~In the event a functions trip setpoint is found non-conservative with respect to the Allowable Value, or the sensors, signal processors, Actuation Signal Voting processors, or actuation devices are found inoperable, then all affected functions provided by that division must be declared inoperable, and the unit must enter any applicable Condition for the particular protection Function affected.~~

When the number of inoperable sensors or signal processors in a reactor trip or ESF function exceeds that specified in any related Condition, redundancy is lost and actions must be taken to restore the required redundancy.

A Note has been added to the ACTIONS. The Note has been added to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each PS sensor, manual actuation switch, signal processor, and actuation device. The Completion Times of each inoperable sensor, manual actuation switch, signal processor, and actuation device will be tracked separately, starting from the time the Condition was entered for that sensor, manual actuation switch, signal processor, and actuation device.

A.1 and A.2

Condition A applies to the failure of one or more sensors. Condition A.1 applies only to the RTCB Position Indication sensors. If one or more of these sensors is inoperable, the inoperable sensor(s) must be placed in the tripped condition in 1 hour. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator action. Condition A.2 applies to all other PS sensors. If one or more of these sensors is inoperable, the inoperable sensor is placed in lockout in 4 hours. The 4 hour allotted timeframe is sufficient to allow the operator to take all appropriate actions for the failed sensor and still ensures that the risk involved in operating with the failure is acceptable.

B.1

Condition B applies to the failure of one or more manual actuation switches. In this condition, the minimum functional capability for manual actuation may not maintained. Restoring the manual initiation capability to OPERABLE status within 48 hours is reasonable considering the availability of automatic actuation, the low probability of an AOO or postulated accident occurring during this time, and the time necessary for repairs.

BASES

ACTIONS (continued)

C.1 and C.2

Condition C applies to one or more APUs inoperable due to the ~~LTSP~~ Setpoint Control Program requirements for one or more Trip/Actuation Functions not met. In this condition, the hardware is still functional. The sensors have been calibrated and the ADOTs and SOTs have checked the function from sensor to actuation device. The manual actuation capability would be unaffected. If the associated ~~inoperability affects the~~ Setpoint Control Program requirements are not met ~~LTSP~~ for either the EDG Start on Degraded Grid Voltage or the EDG Start on LOOP (Trip/Actuation Functions B.10.a or B.10.b), Required Action C.1 directs entry into the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown." The Completion Time of 1 hour is a reasonable time to allow the operator to diagnose and potentially correct the issue that caused the noncompliance with the associated Setpoint Control Program requirements ~~inoperability~~ prior to entering LCO 3.8.1 or LCO 3.8.2. Restoring compliance with the Setpoint Control Program requirements ~~LTSP to OPERABLE status~~ within 24 hours for all other Trip/Actuation Functions is a reasonable timeframe considering the time necessary to change the setpoint parameter, load corrected software, or replace the unit. If ~~the~~ compliance with the Setpoint Control Program requirements~~LTSP~~ cannot be restored ~~to OPERABLE status~~, the associated Trip/Actuation Function must be placed in lockout in the associated APU.

D.1 and D.2

Condition D applies to one or more signal processors inoperable for reasons other than Condition C. If the inoperability affects the APU associated with the EDG Start on Degraded Grid Voltage or the EDG Start on LOOP (Trip/Actuation Functions B.10.a or B.10.b), Required Action D.1 directs entry into the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown." The Completion Time of 1 hour is a reasonable time to allow the operator to diagnose and potentially correct the issue that caused the inoperability prior to entering LCO 3.8.1 or LCO 3.8.2. Restoring the Signal processor to OPERABLE status within 4 hours for all other Trip/Actuation Functions is a reasonable timeframe considering the time necessary to restore the signal processor to OPERABLE status. If the signal processor cannot be restored to OPERABLE status, the signal processor must be placed in lockout.

BASES

ACTIONS (continued)

E.1

Condition E applies to the RCP Bus and Trip Breakers, RTCBs, and Reactor Trip Contactors. With one ore more actuation devices inoperable, the actuation device must be restored to OPERABLE status within 48 hours. The Completion Time of 48 hours is reasonable considering that there are two automatic actuation divisions and the low probability of an event occurring during this interval.

F.1

If the Required Action and associated Completion Time of Condition A, B, C, D, or E or if the minimum functional capability (the value where the supported functions would not actuate during an AOO or postulated event coupled with a single failure) of the sensors, manual actuation switches, signal processors or actuation devices specified in Table 3.3.1-1 are not met, then the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE and any other specified actions must be taken.. The applicable Condition referenced in the table is sensor, manual actuation switch, signal processor, actuation device, and MODE dependent. Condition F is entered to provide for transfer to the appropriate subsequent Condition. Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.1-1.

G.1

If Table 3.3.1-1 directs entry into Condition G, the unit must be brought to a condition in which the Low-Low RCS Loop Flow Rate in One Loop function (Trip/Actuation Function A.6.a) is not required to be OPERABLE. The allowed Completion Time of 2 hours is reasonable, based on operating experience, to reduce THERMAL POWER from full power to less than 70% in an orderly manner and without challenging unit systems.

H.1

If Table 3.3.1-1 directs entry into Condition H, the unit must be brought to a condition in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reduce THERMAL POWER from full power to less than 10% in an orderly manner and without challenging unit systems.

ACTIONS (continued)

I.1

If Table 3.3.1-1 directs entry into Condition I, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 8 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging unit systems.

J.1

If Table 3.3.1-1 directs entry into Condition J, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

K.1 and K.2

If Table 3.3.1-1 directs entry into Condition K, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and open the reactor trip breakers without challenging unit systems.

L.1 and L.2

If Table 3.3.1-1 directs entry into Condition L, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and then reduce the pressurizer pressure to less than 2005 psia without challenging unit systems.

BASES

ACTIONS (continued)

M.1 and M.2

If Table 3.3.1-1 directs entry into Condition M, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours to reach MODE 3 and 12 hours to reach MODE 4 is reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

N.1 and N.2

If Table 3.3.1-1 directs entry into Condition N, the unit must be brought to a MODE in which the supported reactor trips or ESF functions are not required to be OPERABLE. The allowed Completion Time of 6 hours to reach MODE 3 and 36 hours to reach MODE 5 is reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

O.1

If Table 3.3.1-1 directs entry into Condition O, the Conditions specified in LCO 3.8.1, "AC Sources - Operating," or LCO 3.8.2, "AC Sources - Shutdown," for the EDG made inoperable by failure of the 6.9 kV Bus Voltage sensors are required to be entered immediately. The actions of those LCOs provide adequate compensatory actions to assure unit safety.

P.1

If Table 3.3.1-1 directs entry into Condition P, the associated CVCS isolation valves are immediately declared inoperable. The actions of LCO 3.4.9, "Pressurizer," provide adequate compensatory actions to assure unit safety.

Q.1

If Table 3.3.1-1 directs entry into Condition Q, the associated PSRVs are immediately declared inoperable. The actions of LCO 3.4.10, "Pressurizer Safety Relief Valves," provide adequate compensatory actions to assure unit safety.

BASES

ACTIONS (continued)

R.1

If Table 3.3.1-1 directs entry into Condition R, both Control Room Emergency Filtration trains are immediately declared inoperable. The actions of LCO 3.7.10, "Control Room Emergency Filtration (CREF)," provide adequate compensatory actions to assure unit safety.

S.1

If Table 3.3.1-1 directs entry into Condition S, the manual Reactor Trip switches are inoperable. If the switches cannot be returned to OPERABLE status within one hour, actions must be taken to ensure all RCCAs are inserted and the reactor must be placed in a condition where the RCCA can not be withdrawn. This is accomplished by opening the reactor trip breakers. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator action.

T.1

If Table 3.3.1-1 directs entry into Condition T, the associated ALUs must be immediately declared inoperable. If the ALUs cannot be returned to OPERABLE status within one hour, actions must be taken to ensure all RCCAs are inserted and the reactor must be placed in a condition where the RCCA can not be withdrawn. This is accomplished by opening the reactor trip breakers. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator action.

SURVEILLANCE REQUIREMENTS

The SRs for any particular PS sensor, manual actuation switch, signal processor, or actuation device are found in the SR column of Table 3.3.1-1 for that sensor, manual actuation switch, signal processor, or actuation device.

~~REVIEWER'S NOTE~~

~~In order for a plant to take credit for topical reports as the basis for justifying Frequencies, topical reports must be supported by an NRC staff SER that establishes the acceptability of each topical report for that unit.~~

~~BASES~~

~~SURVEILLANCE REQUIREMENTS (continued)~~

~~------------------------------- REVIEWER'S NOTE ---------------------------------~~
~~The Notes in Table 3.3.1-1 requiring reset of the division to a predefined as-left tolerance and the verification of the as-found tolerance are only associated with SL-LSSS values. Therefore, the Notes are placed at the top of the LTSP column in the Table and applied to all Functions with LTSPs in the table. The Notes may be applied to specific SRs for the associated functions in the SR column only.~~
~~---~~

~~------------------------------- REVIEWER'S NOTE ---------------------------------~~
~~Notes b and c are applied to the setpoint verification Surveillances for all SL-LSSS Functions unless one or more of the following exclusions apply:~~

~~1.Notes b and c are not applied to SL-LSSS Functions which utilize mechanical components to sense the trip setpoint or to manual initiation circuits (the latter are not explicitly modeled in the accident analysis). Examples of mechanical components are limit switches, float switches, proximity detectors, manual actuation switches, and other such devices that are normally only checked on a "go/no go" basis. Note 1 requires a comparison of the periodic surveillance requirement results to provide an indication of Trip/Actuation Function (or individual device) performance. This comparison is not valid for most mechanical components. While it is possible to verify that a limit switch functions at a point of travel, a change in the surveillance result probably indicates that the switch has moved, not that the input/output relationship has changed. Therefore, a comparison of surveillance requirement results would not provide an indication of the Trip/Actuation Function or component performance.~~

~~2.Notes b and c are not applied to Technical Specifications associated with mechanically operated safety relief valves. The performance of these components is already controlled (i.e., trended with as-left and as-found limits) under the ASME Section XI testing program.~~

~~3.Notes b and c are not applied to SL-LSSS Functions and Surveillances which test only digital components. For purely digital components, such as actuation logic circuits and associated relays, there is no expected change in result between surveillance performances other than measurement and test errors (M&TE) and, therefore, comparison of Surveillance results does not provide an indication of Trip/Actuation Function or component performance.~~

~~BASES~~

~~SURVEILLANCE REQUIREMENTS (continued)~~

~~An evaluation of the potential SL-LSSS Functions resulted in Notes b and c being applied to the Functions shown in the TS markups. Each licensee proposing to fully adopt this TSTF must review the potential SL-LSSS Functions to identify which of the identified functions are SL-LSSS according to the definition of SL-LSSS and their plant specific safety analysis. The two TSTF Notes are not required to be applied to any of the listed Functions which meet any of the exclusion criteria or are not SL-LSSS based on the plant specific design and analysis.~~

~~The Limiting Trip Setpoint column for reactor trip functions is modified by two Footnotes as identified in Table 3.3.1-2. The selected Functions are those Functions that are LSSS for protection system instrument functions that protect reactor core or RCS pressure boundary SLs. Some components (e.g., mechanical devices which have an on or off output or an open/close position such as limit switches, float switches, and proximity detectors) are not calibrated in the traditional sense and do not have as-left or as-found conditions that would indicate drift of the component setpoint. These devices are considered not trendable and the requirements of the Notes are not required to be applied to the mechanical portion of the functions. Where a non-trendable component provides signal input to other Trip/Actuation Function components that can be trended, the remaining components must be evaluated in accordance with the Notes.~~

~~The first Note requires evaluation of Trip/Actuation Function's performance for the condition where the as-found setting for the setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. For digital channel components, the as-found tolerance may be identical to the as-left tolerance since drift may not be an expected error. In these cases a Trip/Actuation Function's as-found value outside the as-left condition may be cause for component assessment. Evaluation of instrument performance will verify that the instrument will continue to behave in accordance with design-basis assumptions. The purpose of the assessment is to ensure confidence in the instrument performance prior to returning the instrument to service. These conditions will also be identified in the Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition for continued OPERABILITY.~~

~~The second Footnote requires that the as-left setting for the instrument be returned to within the as-left tolerance of the LTSP. Where a setpoint more conservative than the LTSP is used in the plant surveillance procedures, the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~sufficient margin to the SL and/or Analytical Limit is maintained. If the as-left instrument setting cannot be returned to a setting within the as-left tolerance, then the Trip/Actuation Function shall be declared inoperable. The second Note also requires that the LTSP and the methodologies for calculating the as-left and the as-found tolerances be in a document controlled under 10 CFR 50.59.~~

The digital PS provides continual online automatic monitoring of each of the input signal in each division, perform software limit checking (signal online validation) against required acceptance criteria, and provide hardware functional validation so that a division check is continuously being performed. If any PS input signal is identified to be in a failure status, this condition is alarmed in the Control Room. As such, a periodic "channel check" is no longer necessary.

SR 3.3.1.1

SR 3.3.1.1~~2~~ compares the calorimetric heat balance calculation to the power range division output every 24 hours. If the calorimetric heat balance calculation results exceed the power range division output by more than 2% RTP, the power range division is not declared inoperable, but must be adjusted. The power range division output shall be adjusted consistent with the calorimetric heat balance calculation results if the calorimetric calculation exceed the power range division output by more than + 2% RTP. If the power range division output cannot be properly adjusted, the division I is declared inoperable.

If the calorimetric is performed at part power (< 70% RTP), adjusting the power range division indication in the increasing power direction will assure a reactor trip below the safety analysis limit (< 11% RTP). Making no adjustment to the power range division 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 is less than the power range division output. To provide close agreement between indicated power and to preserve operating margin, the power range divisions are normally adjusted when operating at or near full power during steady-state conditions. However, discretion must be exercised if the power range division output is adjusted in the decreasing power direction due to a part power calorimetric (< 70% RTP). This action may introduce a non-conservative bias at higher power levels. The cause of

SURVEILLANCE REQUIREMENTS (continued)

the potential non-conservative bias is the decreased accuracy of the calorimetric at reduced power conditions. The primary error contributor to the instrument uncertainty for a secondary side power calorimetric measurement is the feedwater flow measurement, which is typically a delta P measurement across a feedwater venturi. While the measurement uncertainty remains constant in delta P as power decreases, when translated into flow, the uncertainty increases as a square term. Thus a 1% flow error at 100% power can approach a 10% flow error at 30% RTP even though the delta P error has not changed. An evaluation of extended operation at part power conditions would conclude that it is prudent to administratively adjust the setpoint of the High Neutron Flux Rate of Change when: 1) the power range division output is adjusted in the decreasing power direction due to a part power calorimetric below 70% RTP; or 2) for a post refueling startup. The evaluation of extended operation at part power conditions would also conclude that the potential need to adjust the indication of the High Neutron Flux Rate of Change in the decreasing power direction is quite small, primarily to address operation in the intermediate range about 10% RTP to allow enabling of the High Neutron Flux Rate of Change reactor trips. Before the High Neutron Flux Rate of Change setpoint is reset, the power range division adjustment must be confirmed based on a calorimetric performed at ≥ 70% RTP.

The Note clarifies that 12 hours are allowed for performing the first Surveillance after reaching 20% RTP. A power level of 20% RTP is chosen based on plant stability, (i.e., automatic rod control capability and turbine generator synchronized to the grid). 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 division output of more than +2% RTP is not expected in any 24 hour period.

SR 3.3.1.2

Space- and time- dependent power density distribution of the U.S. EPR is accurately assessed using the SPNDs inside the core. For neutron flux measurement, incore neutron detectors are more accurate than excore neutron detectors. CALIBRATION of SPND instrumentation is performed to compensate for a decrease in SPND sensitivity during the fuel cycle and to account for peak power density factor change over the fuel cycle. The Aeroball Measurement System (AMS) assists in generating the measured relative neutron flux density in the core, which is used in

BASES

SURVEILLANCE REQUIREMENTS (continued)

conjunction with the predicted power distribution based on actual core operation to calibrate the incore SPND instrumentation. Because both the power-to-signal ratio of an SPND and the reference power distribution change with core burnup, SPND signals are matched to reference signals provided by the AMS every 15 EFPD (Ref. 7).

The Note clarifies that 12 hours are allowed for performing the first Surveillance after reaching 20% RTP. A power level of 20% RTP is chosen based on plant stability, (i.e., automatic rod control capability and turbine generator synchronized to the grid).

SR 3.3.1.3

SR 3.3.1.3 is the performance of a ADOT every 31 days. This test shall verify OPERABILITY by actuation of the Reactor Trip Circuit Breakers and Reactor Trip Contactors. The ADOT may be performed by means of any series of sequential, overlapping, or total steps.

SR 3.3.1.4

The online boron meters are a half shell design and are not in contact with the reactor coolant. The concentration of boron is measured by using the neutron absorption effect of B^{10}. The boron concentration is calculated using the measured count rate. To improve the accuracy of the measurement, the temperature of the reactor coolant at the measuring point is used to adjust the boron concentration. The temperature instruments are not included as part of this Surveillance. The frequency of the boron meter CALIBRATION is conservative considering instrument reliability.

Specification 5.5.18.a requires that the Limiting Trip Setpoint (LTSP), Allowable Value (AV), as-found tolerance (AFT), and the as-left tolerance (ALT), as well as the methodology for calculating these be in the Setpoint Control Program (SCP).

The SCP provides requirements for the calibration reset and evaluation of the performance of required divisions. As indicated in Specification 5.5.18.c.1 evaluation of division performance is required for the condition where the "as-found" setting for the division is outside its AFT, but conservative with respect to the AV. Evaluation of division performance will verify that the instrument will continue to behave in accordance with design-basis assumptions. The purpose of the assessment is to ensure confidence in the instrument performance prior to returning the instrument to service. These divisions will also be identified in the Corrective Action

BASES

SURVEILLANCE REQUIREMENTS (continued)

Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition for OPERABILITY. For digital division components and Functions whose instruments are mechanical devices (e.g., devices which have an "on" or "off" output or an open/close position such as limit switches, float switches, and proximity detectors), the AFT may be identical to the ALT because drift may not be an expected error.

As indicated in Specification 5.5.18.c.2, the as-left setting for the instrument is required to be returned to within the ALT around the LTSP. Where a setpoint more conservative than the LTSP is used in plant surveillance procedures, the ALT and AFT, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the AV is maintained. If the as-left instrument setting cannot be returned to a setting within the ALT, then the instrument division shall be declared inoperable.

SR 3.3.1.5

A SOT on each required reactor trip actuation device is performed every 24 months to ensure the devices will perform their intended function when needed. A SOT shall be the injection of a simulated or actual signal into the division as close to the sensor as practicable to verify OPERABILITY of all devices in the division required for division OPERABILITY. The SOT shall include the verification of the accuracy and time constants of the analog input modules. Any setpoint adjustment shall be consistent with the assumptions of the current setpoint methodology as required by the SCP.

The maximum permissible response time for analog input modules is prescribed by the process engineering of the specific application. Thus for each applicable PS function, the limiting response times will be shown to be consistent with the safety requirements.

The response time testing is performed in overlapping steps:

- Verification of time constants of the input divisions during input module tests, and

- Verification of the signal propagation time within the digital system.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The response time of the analog input divisions are tested periodically by injection of test signals in the input circuits. For this purpose, an external test computer is temporarily connected to the I&C system via permanently installed test plugs. While the input from the process is deactivated (by switching off the associated division(s) power supply), a binary input is provided to the data acquisition computers. The signal distribution to other computers is designed in the application software in the same way as for the normal measuring signals. Separate outputs are provided in the voting computers for each path. During the response time tests, the test machine connected to the I&C system generates a start signal and measures the reaction time of each signal path separately to verify that it does not exceed the worst case conditions specified for the specific system configuration. The measurements are performed a number of times to determine the statistical characteristics of each signal path.

The SOT may be performed by means of any series of sequential, overlapping, or total steps.

SR 3.3.1.6

A CALIBRATION of each PS sensor (except neutron detectors) every 24 months ensures that each instrument division is reading accurately and within tolerance. A CALIBRATION shall be the adjustment, as necessary, of the sensor output such that it responds within the necessary range and accuracy to known values of the parameter that the sensor monitors. The CALIBRATION shall encompass all devices in the division required for sensor OPERABILITY. CALIBRATION of instrument divisions with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and normal CALIBRATION of the remaining adjustable devices in the division. The CALIBRATION may be performed by means of any series of sequential, overlapping, or total steps.

Specification 5.5.18.a requires that the Limiting Trip Setpoint (LTSP), Allowable Value (AV), as-found tolerance (AFT), and the as-left tolerance (ALT), as well as the methodology for calculating these be in the Setpoint Control Program (SCP).

The SCP provides requirements for the calibration reset and evaluation of the performance of required divisions. As indicated in Specification 5.5.18.c.1 evaluation of division performance is required for the condition where the "as-found" setting for the division is outside its AFT, but conservative with respect to the AV. Evaluation of division performance will verify that the instrument will continue to behave in accordance with design-basis assumptions. The purpose of the assessment is to ensure

SURVEILLANCE REQUIREMENTS (continued)

confidence in the instrument performance prior to returning the instrument to service. These divisions will also be identified in the Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition for OPERABILITY. For digital division components and Functions whose instruments are mechanical devices (e.g., devices which have an "on" or "off" output or an open/close position such as limit switches, float switches, and proximity detectors), the AFT may be identical to the ALT because drift may not be an expected error.

As indicated in Specification 5.5.18.c.2, the as-left setting for the instrument is required to be returned to within the ALT around the LTSP. Where a setpoint more conservative than the LTSP is used in plant surveillance procedures, the ALT and AFT, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the AV is maintained. If the as-left instrument setting cannot be returned to a setting within the ALT, then the instrument division shall be declared inoperable.

SR 3.3.1.7

The features of continuous self-monitoring of the PS system are described in Reference 8. Additional tests, which require the processor to be inoperable are not normally performed during operation. These EXTENDED SELF TESTS are performed at start-up of a computer each cycle. The startup sequence is as follows:

- Hardware basic test using the internal diagnosis monitor,
- Start-up self test of the operating system, and
- Switch over to normal operation after approximately two minutes.

Additional information is provided in Section 3 of Reference 8.

SR 3.3.1.8

SR 3.3.1.8 is the performance of a ADOT every 31 days. This test shall verify OPERABILITY by actuation of the RCP Bus and Trip Breakers. The ADOT may be performed by means of any series of sequential, overlapping, or total steps.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.9

SR 3.3.1.9 verifies that the ~~Limiting Trip S~~setpoint and p~~P~~ermissive values have been properly loaded into the applicable APU.

REFERENCES

1. ANP-10275P, Revision 0, U.S. EPR Instrument Setpoint Methodology Topical Report, March 2007.

2. 10 CFR 100.

3. 10 CFR 50, Appendix A, GDC 21.

4. ANP-10287, Incore Trip Setpoint and Transient Methodology for U.S. EPR, November 2007.

5. FSAR Chapter 15.

6. 10 CFR 50.49.

7. ANP-10271P, Revision 0, US EPR Nuclear Incore Instrumentation Systems Report, December 2006.

8. EMF-2341(P), Revision 1, Generic Strategy for Periodic Surveillance Testing of TELEPERM XS System in U.S. Nuclear Generating Stations, March 2000.

Table B 3.3.1-1 (page 1 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
A. REACTOR TRIPS						
1. Low Departure from Nucleate Boiling Ratio (DNBR) a. Low DNBR d. High Quality	≥ 10% RTP	3	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit

Table B 3.3.1-1 (page 2 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
1. Low Departure from Nucleate Boiling Ratio (DNBR) b. Low DNBR and (Imbalance or Rod Drop) c. Variable Low DNBR and Rod Drop e. High Quality and Imbalance or Rod Drop	≥ 10% RTP	3	A total of 65 RCCA Position Indicators in any of the four divisions A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One RCCA Unit per division with a required OPERABLE RCCA position indicator One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 65 RCCA Position Indicators in any of the four divisions A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One RCCA Unit per division with a required OPERABLE RCCA position indicator One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 65 RCCA Position Indicators in any of the four divisions A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One RCCA Unit per division with a required OPERABLE RCCA position indicator One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 65 RCCA Position Indicators in any of the four divisions A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / One RCCA Unit per division with a required OPERABLE RCCA position indicator One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit

Table B 3.3.1-1 (page 3 of 7)
Protection System (PS) Functional Dependencies

	TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
2.	High Linear Power Density	≥ 10% RTP	3	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of 51 Self-Powered Neutron Detectors (SPND) in any of the four divisions / One Remote Acquisition Unit per division with a required OPERABLE SPND Acquisition and Processing Unit
4.	High Core Power Level	1,2[(a)]	3	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit

(a) ≥ 10-5% power on the intermediate range detectors.

Table B 3.3.1-1 (page 4 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
5. Low Saturation Margin	1,2[(a)]	3	Cold Leg Temperature (WR) Hot Leg Temperature (NR) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit

(a) ≥ 10-5% power on the intermediate range detectors.

Table B 3.3.1-1 (page 5 of 7)
Protection System (PS) Functional Dependencies

	TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
B.	ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) SIGNALS						
2.e.	Main Feedwater / Startup and Shutdown Feedwater Isolation on Steam Generator Level High for Period of Time (Affected Steam Generators)	1,2[(b)],3[(b)]	3	Steam Generator Level (NR) Reactor Trip Circuit Breaker Position Indication / Acquisition and Processing Unit	Steam Generator Level (NR) Reactor Trip Circuit Breaker Position Indication / Acquisition and Processing Unit	Steam Generator Level (NR) Reactor Trip Circuit Breaker Position Indication / Acquisition and Processing Unit	Steam Generator Level (NR) Reactor Trip Circuit Breaker Position Indication / Acquisition and Processing Unit
3.b.	ESF - Safety Injection System (SIS) Actuation on Low Delta P_{sat}	3[(c)]	3	Hot Leg Pressure (WR) Hot Leg Temperature (WR) / Acquisition and Processing Unit	Hot Leg Pressure (WR) Hot Leg Temperature (WR) / Acquisition and Processing Unit	Hot Leg Pressure (WR) Hot Leg Temperature (WR) / Acquisition and Processing Unit	Hot Leg Pressure (WR) Hot Leg Temperature (WR) / Acquisition and Processing Unit

(b) Except when all MFW low load isolation valves are closed.

(c) When Trip/Actuation Function B.3.a, SIS Actuation on Low Pressurizer Pressure, is disabled.

Table B 3.3.1-1 (page 6 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
4. ESF - Reactor Coolant Pump (RCP) Trip on Low Delta P across RCP with Safety Injection System Actuation	1,2,3	3	RCP Current (2 of 3) RCP Delta P (1 of 2) / Acquisition and Processing Unit	RCP Current (2 of 3) RCP Delta P (1 of 2) / Acquisition and Processing Unit	RCP Current (2 of 3) RCP Delta P (1 of 2) / Acquisition and Processing Unit	RCP Current (2 of 3) RCP Delta P (1 of 2) / Acquisition and Processing Unit
11a. ESF - Chemical and Volume Control System (CVCS) Charging Line Isolation on High-High Pressurizer Level	1,2,3	3	Pressurizer Level / Acquisition and Processing Unit	Pressurizer Level / Acquisition and Processing Unit	Pressurizer Level / Acquisition and Processing Unit	Pressurizer Level / Acquisition and Processing Unit
	1,2,3	2	Actuation Logic Unit (1 of 2)			Actuation Logic Unit (1 of 2)

Table B 3.3.1-1 (page 7 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
11b. ESF - Chemical and Volume Control System (CVCS) Charging Line Isolation on Anti-Dilution Mitigation (ADM) at Shutdown Condition (RCP not operating)	5[(d)],6	3	Boron Concentration Boron Temperature / Acquisition and Processing Unit	Boron Concentration Boron Temperature / Acquisition and Processing Unit	Boron Concentration Boron Temperature / Acquisition and Processing Unit	Boron Concentration Boron Temperature / Acquisition and Processing Unit
	5[(d)],6	2	Actuation Logic Unit (1 of 2)			Actuation Logic Unit (1 of 2)
11c. ESF - Chemical and Volume Control System (CVCS) Charging Line Isolation on ADM at Standard Shutdown Conditions	3,4[(e)],5[(e)]	3	Boron Concentration Boron Temperature Chemical and Volume Control System (CVCS) Charging Line Flow Cold Leg Temperature (WR) / Acquisition and Processing Unit	Boron Concentration Boron Temperature Chemical and Volume Control System (CVCS) Charging Line Flow Cold Leg Temperature (WR) / Acquisition and Processing Unit	Boron Concentration Boron Temperature Chemical and Volume Control System (CVCS) Charging Line Flow Cold Leg Temperature (WR) / Acquisition and Processing Unit	Boron Concentration Boron Temperature Chemical and Volume Control System (CVCS) Charging Line Flow Cold Leg Temperature (WR) / Acquisition and Processing Unit
	3,4[(e)],5[(e)]	2	Actuation Logic Unit (1 of 2)			Actuation Logic Unit (1 of 2)

(d) With two or less RCPs in operation.

(e) With three or more RCPs in operation.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as containment post tensioning surveillance, or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and ≤ 0.75 L_a for overall Type A leakage. At all other times between required Containment Leakage Rate Testing Program leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of 1.0 L_a. At 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis.

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

~~--------------------------------REVIEWER' NOTE---------------------------------~~
~~Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.~~
~~---~~

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Post Tensioning Surveillance Program. Testing and Frequency are in accordance with the ASME Code, Section III, Division 2, 2004 (Ref. 4).

B 3.7 PLANT SYSTEMS

B 3.7.8 Essential Service Water (ESW) System

BASES

BACKGROUND

The ESW System provides a heat sink for the removal of process and operating heat from safety related components during an anticipated operational occurrence (AOO) or postulated accident. During normal operation, and a normal shutdown, the ESW System also provides this function for the associated safety related and nonsafety related systems. The safety related function is covered by this LCO.

The ESW System consists of four separate safety related, cooling water trains. Each train consists of one mechanical draft cooling tower, associated basin, pump, piping, valving, instrumentation, and mechanical filtration. Each safety related 2-cell seismic Category I mechanical draft cooling tower rejects energy from the ESW fluid to the ambient and returns the cooled fluid to the ESW cooling tower basin, from which the ESW pumps take suction. Each ESW cooling tower basin is sized for 3 days of post loss of coolant accident (LOCA) operation and ensures adequate volume for the required net positive suction head (NPSH) for the associated ESW pump. Post LOCA evaporative losses are replenished by a safety related seismic Category I source of makeup water. The train associated safety related make-up source delivers water to each basin at ≥ 300 gpm to maintain the NPSH for the ESW pump for up to 30 days following a LOCA. The system pumps and valves are remote and manually aligned, except in the unlikely event of a LOCA or loss of offsite power. The pumps aligned to the critical loops are automatically started upon receipt of a safety injection signal, and all essential valves are aligned to their post accident positions.

The mechanical draft cooling towers and basins are safety related, seismic Category I structures sized to provide heat dissipation for safe shutdown following an accident. The cooling tower is protected from tornado missiles.

~~[The seismic Category 1 makeup necessary to support 30 days of post accident mitigation is site specific and details are to be provided by the Combined License applicant]~~The seismic Category 1 emergency makeup water supply, to the ESW cooling tower basins, necessary to support 30 days of post accident mitigation is provided by the safety-related Ultimate Heat Sink (UHS) Makeup Water System that draws water from the Chesapeake Bay. Chesapeake Bay water enters the UHS Makeup Water Intake Structure through an intake channel shared the Circulating Water System Makeup Intake Structure. The UHS Makeup Water Intake Structure houses four independent UHS Makeup Water System trains, one for each ESW division. Each train has one pump, a discharge check valve, and a pump

BASES

BACKGROUND (continued)

discharge isolation motor operated valve, all housed in the UHS Makeup Water Intake Structure, plus the buried piping running up to and into the ESW pumphouse at the ESW cooling tower basin. Each UHS Makeup Water System pump is rated at 750 gpm.

Additional information about the design and operation of the ESW System along with a list of the components served, is presented in FSAR Section 9.2.1 (Ref. 1). The principal safety related functions of the ESW System is the removal of decay heat from the reactor and reactor coolant pump thermal barrier cooling via the Component Cooling Water (CCW) System and removal of operational heat from the emergency diesel generator (EDG).

APPLICABLE SAFETY ANALYSES

The design basis of the ESW System is for two ESW trains, in conjunction with the CCW System, to remove core decay heat and support containment cooling following a design basis LOCA as discussed in FSAR Section 6.2 (Ref. 2). This maintains the In-containment Water Storage Tank fluid within acceptable limits following a LOCA as it is supplied to the Reactor Coolant System by the Emergency Core Cooling System pumps. The ESW System also provides cooling to the train EDG during an anticipated operational occurrence (AOO) or postulated accident.

The ESW System, in conjunction with the CCW System, also cools the unit from residual heat removal (RHR), as discussed in FSAR Section 5.4.7 (Ref. 3), entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of CCW and RHR loops that are operating. Two ESW trains are sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum ESW System temperature of 95°F occurring simultaneously with maximum heat loads on the system.

Each ESW basin is sized for 3 days of post LOCA operation without requiring makeup. ESW basin makeup is required to maintain NPSH for the ESW pumps beyond 3 days. This volume of water is assumed to be at ≤ 90°F during normal plant operation to prevent exceeding the maximum ESW temperature during a LOCA.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis, which include worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and worst case single active failure. The ESW cooling tower and basin is designed in accordance with Regulatory Guide 1.27 (Ref. 4), which requires a 30 day supply of cooling water in the ESW basin, or equivalent make-up.

The ESW System satisfies Criterion 2 and 3 of 10 CFR 50.36(d)(2)(ii).

LCO

The ESW System consists of four trains. Four ESW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads.

An ESW train is considered OPERABLE when two cooling tower fans, pump, associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE and the ESW basin contains ≥ 27.2 feet of water at ≤ 90°F with capability from makeup from ~~Operable~~ OPERABLE source. ~~[COL applicant to provide definition of OPERABLE makeup source.]~~An OPERABLE emergency makeup water source consists of one OPERABLE train of the UHS Makeup Water System capable of providing makeup water to its associated ESW cooling tower basin. Each UHS Makeup Water System train includes a pump, valves, piping, instruments and controls to ensure the transfer of the required supply of water from the Chesapeake Bay to its associated ESW cooling tower.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ESW System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ESW System and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ESW System are determined by the systems it supports.

BASES

ACTIONS

The actions have two Notes added. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," should be entered if an inoperable ESW train results in an inoperable EDG. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable ESW train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

A.1

If one ESW train is inoperable, action must be taken to restore OPERABLE status within 120 days. In this condition, the remaining OPERABLE ESW trains are adequate to perform the heat removal function.

The 120 day Completion Time to restore an ESW train to OPERABLE is reasonable since its operation is not assumed in the safety analysis to mitigate the consequences of postulated accidents or AOOs, it provides a reasonable time for repairs, and the low probability of a postulated accident or AOO occurring during this period.

B.1

If two ESW trains are inoperable, action must be taken to restore one to OPERABLE status within 72 hours. In this condition, the two remaining OPERABLE ESW train are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in one of the OPERABLE ESW trains could result in loss of ESW System function.

The 72 hour Completion Time is based on the redundant capabilities afforded by the two OPERABLE trains, and the low probability of a postulated accident occurring during this time period.

C.1 and C.2

If an ESW train cannot be restored to OPERABLE status within the associated Completion Time, 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 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR verifies that adequate short term (3 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the ESW pumps during the first 3 days post LOCA. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the ESW basin water level is ≥ 27.2 feet from the bottom of the basin.

SR 3.7.8.2

This SR verifies that the ESW System is available to cool the CCW System and EDG to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a postulated accident. With water temperature of the ESW basin ≤ ~~[~~90~~]~~°F, the design basis assumption associated with initial ESW temperature are bounded. With the water temperature of the ESW basin > 90°F, long term cooling capability of the Emergency Core Cooling System (ECCS) loads and Diesel Generators (DGs) may be affected. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.8.3

This SR is modified by a Note indicating that the isolation of the ESW components or systems may render those components inoperable, but does not affect the OPERABILITY of the ESW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the ESW flow path provides assurance that the proper flow paths exist for ESW System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.4

Operating each cooling tower fan for ≥ 15 minutes in all speed settings verifies that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration, can be detected for corrective action. The 31 day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the ESW cooling tower fans occurring between surveillances.

SR 3.7.8.5

This SR verifies proper automatic operation of the ESW valves on an actual or simulated actuation signal. The ESW System is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.6

This SR verifies proper automatic operation of the ESW pumps and cooling tower fans on an actual or simulated actuation signal. The ESW System is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.7

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified makeup flowrate ensures that sufficient NPSH can be maintained to operate the ESW pumps following the first 3 days post LOCA. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint. This SR verifies that the ESW makeup flowrate is $\geq$ 300 gpm.

REFERENCES

1. FSAR Section 9.2.1.

2. FSAR Section 6.2.

3. FSAR Section 5.4.7.

4. Regulatory Guide 1.27.

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Emergency Filtration (CREF)

BASES

BACKGROUND

The CREF provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity~~,~~ ~~[[hazardous chemicals]],~~ or smoke.

The CREF consists of two 100% capacity iodine filtration trains which operate when radioactive contamination is detected at the site or inside the control room envelope (CRE) area. The iodine filtration train is a bypass path of the fresh air intake train for the Control Room Air Conditioning System (CRACS) normal air supply. The air from CRE can also be recirculated through the CREF Iodine Filtration trains. The iodine filtration trains are provided as bypass lines on two of the four normal CRACS air intake trains; other two CRACS intake trains do not have the bypass iodine filtration trains. During an emergency, the fresh outside air and recirculated air are directed through air intake motorized damper and electric heater through the CREF Iodine Filtration train. Each iodine filtration train consists of motorized damper, electric heater, prefilter, upstream HEPA filter, an activated carbon iodine filter, downstream HEPA filter, booster fan, and manual isolation damper. The filtered and clean air is then directed through one or both CRACS normal 75% capacity air conditioning train. Each air conditioning train consists of volume control manual damper, cooling coil, moisture separator, fan suction and discharge silencers, supply air fan, HEPA filter, steam humidifier, non-return damper, volume control electric damper, and fire dampers. The conditioned and clean air is then supplied to the CRE areas. Electric heaters are installed in the CRE supply air ducts to maintain individual room temperatures and relative humidity. The exhaust air from the CRE areas is directed through the recirculation air shaft and then recycled either through the iodine filtration trains or CRACS air conditioning trains. The exhaust from kitchen and sanitary areas is separated from the recycle return air and processed separately.

The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and carbon adsorbers. The HEPA filter bank downstream of the carbon iodine filter collects carbon fines and provides backup in case of failure of the upstream HEPA filter bank. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and carbon adsorbers.

BASES

BACKGROUND (continued)

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The CREF train is an emergency system, which may also operate during normal unit operations in the standby mode of operation. Upon receipt of the actuating signal(s), the outside fresh air supply to the CRE is isolated, and the outside air is directed through the CREF train. The CRE ventilation air is recycled through the air conditioning filter trains and/or CREF train.

Actuation of the CREF places the system in ~~either of two separate states~~ ~~(~~the emergency radiation ~~state or toxic gas isolation state) of the emergency~~ mode of operation~~, depending on the initiation signal~~. Actuation of the system to the emergency radiation ~~state of the emergency~~ mode of operation, closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the air within the CRE through the CREF trains. The emergency radiation state also maintains control room pressurization and filtered ventilation of the air supply to the CRE.

Outside makeup air is supplied through the iodine filtration train and added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary. ~~The actions taken in the toxic gas isolation state are the same, except that the signal switches the CREF to an isolation alignment to minimize any outside air from entering the CRE through the CRE boundary.~~

The outside air entering the CRE is continuously monitored by radiation ~~[[and toxic gas]]~~ detectors. One detector output above the setpoint will cause actuation of the emergency radiation state or toxic gas isolation state, as required. ~~The actions of the toxic gas isolation state are more restrictive, and will override the actions of the emergency radiation state.~~

BASES

BACKGROUND (continued)

One CREF operating at a flow rate of < 4000 cfm will pressurize the CRE to ≥ 0.125 inches water gauge relative to all external areas adjacent to the CRE boundary. The CREF operation in maintaining the CRE habitability is discussed in FSAR Section 9.4.1 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across one of the other filter trains. Normally open isolation dampers are arranged in series so the failure of one damper to shut will not result in a breach of isolation. The CREF train components are designed in accordance with Seismic Category I requirements.

The CREF is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a postulated accident without exceeding a 5 rem whole body dose or its equivalent to any part of the body 5 rem total effective does equivalent (TEDE).

APPLICABLE SAFETY ANALYSES

The CREF components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The CREF provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis loss of coolant accident, fission product release presented in Chapter 15 (Ref. 2).

The CREF consists of two 100% capacity iodine filtration trains. Each iodine filtration train can be aligned with one of the two 75% capacity air conditioning trains. There are only two iodine filtration trains since only slow failure modes are assumed and filtration efficiency is checked periodically. Both CREF trains with the associated air conditioning trains are required to be OPERABLE. One CREF train is assumed to be lost to a single failure. The other train provides 100% of the ventilation to the CRE.

The CREF provides protection from smoke ~~[[and hazardous chemicals~~ to the CRE occupants. Reference 3 discusses that the need for protection of CRE occupants following a hazardous chemical release~~]]~~ is not required at CCNPP Unit 3. Reference 4 discusses protection of the CRE occupants and their ability to control the reactor from the control room or from the remote shutdown panels in the event of a smoke challenge.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The worst case single active failure of a component of the CREF, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The CREF satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

In the event of a postulated accident, one iodine filtration train is required to provide an adequate supply of filtered air to the CRE. To ensure that this requirement is met, both CREF trains must be OPERABLE. The basis for this approach is that two trains are required to satisfy all design requirements (i.e., one train is needed to mitigate the event and other train is assumed to have a single active failure). The failure of both iodine filtration trains could result in exceeding a dose of 5 rem whole body or its equivalent to any part of the body 5 rem TEDE in the event of a large radioactive release.

Each CREF train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. A CREF train is OPERABLE when the associated:

a. Fan is OPERABLE;

b. Prefilters, HEPA filters, and carbon adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Heater, ductwork, and dampers are OPERABLE, and air circulation can be maintained.

In order for the CREF trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for postulated accident~~s~~, and that CRE occupants are protected from ~~[[hazardous chemicals and]]~~ smoke.

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design conditions, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be procedurelized, and consist of

.

BASES

LCO (continued)

stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of irradiated fuel assemblies, the CREF trains must be OPERABLE to ensure that the CRE will remain habitable during and following a postulated accident (i.e., LOCA, main steam line break, rod ejection, and fuel handling accident).

In MODE 5 or 6, the CREF is also required to cope with a failure of the Gaseous Waste Processing System.

ACTIONS

A.1

With one CREF train inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the OPERABLE CREF train is adequate to perform the CRE occupant protection function. However, the overall system reliability is reduced. The 7 day Completion Time is based on the low probability of a postulated accident occurring during this time period, and ability of the remaining trains to provide the required capability.

B.1, B.2, and B.3

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of postulated accident consequences (allowed to be up to 5 rem whole body or its equivalent to any part of the body 5 rem TEDE), or inadequate protection of CRE occupants from ~~[[hazardous chemicals or]]~~ smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological ~~[[or chemical]]~~ event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a postulated accident, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of postulated accident

BASES

ACTIONS (continued)

consequences, and that CRE occupants are protected from ~~hazardous chemicals and~~ smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a postulated accident occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a postulated accident. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

<u>C.1 and C.2</u>

In MODE 1, 2, 3, or 4, if any Required Action and Completion Time of Condition A or B cannot be met, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 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.

<u>D.1 and D.2</u>

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CREF train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREF train in the emergency mode. This action ensures that the other train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

~~Required Action D.1 is modified by a Note indicating to place the system in the toxic gas isolation state with outside air isolated.~~

BASES

ACTIONS (continued)

E.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CREF trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the CRE. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

F.1

With both Iodine Filtration trains and associated Air Conditioning trains inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable CRE boundary (i.e., Condition B), the CREF may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Monthly heater operations which dry out any moisture accumulated in the carbon from humidity in the ambient air should be performed. Each Iodine filtration train must be operated for ≥ 15 minutes with the heaters energized. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy.

SR 3.7.10.2

This SR verifies that the required CREF train testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, carbon adsorber efficiency, minimum flow rate, and the physical properties of the activated carbon. Specific test Frequencies and additional information are discussed in detail in the VFTP.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.3

This SR verifies that each CREF train starts and operates on an actual or simulated actuation signal. The Frequency of 24 months is based on industry operating experience and is consistent with the typical refueling cycle.

SR 3.7.10.4

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of postulated accident consequences is no more than 5 rem whole body or its equivalent to any part of the body 5 rem TEDE and the CRE occupants are protected from ~~[[hazardous chemicals and]]~~ smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of postulated accident consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Mitigating actions, or compensatory measures, are discussed in Regulatory Guide 1.196, Section 2.7.3, (Ref. 5) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 6). These compensatory measures may also be used as mitigating measures as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures (Ref. 7). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis postulated accident consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

BASES

REFERENCES

1. FSAR Section 9.4.
2. Chapter 15.
3. FSAR Section 6.4.
4. FSAR Section 9.5.
5. Regulatory Guide 1.196.
6. NEI 99-03, "Control Room Habitability Assessment," March 2003.
7. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2005, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability" (ADAMS Accession No. ML040300694).

BASES

ACTIONS (continued)

B.1

~~-------------------------------------REVIEWER'S NOTE------------------------------~~
~~Adoption of Condition B is dependent on a commitment from the licensee to have guidance available describing compensatory measures to be taken in the event of an intentional and unintentional entry into Condition B.~~
~~--~~

If the safeguard buildings or fuel building boundary is inoperable in MODE 1, 2, 3, or 4, the SBVS trains may not be able to perform their intended functions. Actions must be taken to restore an OPERABLE safeguard buildings and fuel building boundaries within 24 hours. During the period that the safeguard buildings or fuel building boundary is inoperable, appropriate compensatory measures consistent with the intent, as applicable, of GDC 19 and 10 CFR Part 100 ~~should~~ <u>shall</u> be utilized to protect plant personnel from potential hazards such as radioactive contamination,~~[[toxic chemicals,]]~~ smoke, temperature and relative humidity, and physical security. Preplanned measures ~~should~~ <u>shall</u> be available <u>and implemented upon entry into the condition</u> to address these concerns ~~for~~ <u>regardless of whether the entry is</u> intentional ~~and~~ <u>or</u> unintentional entry ~~into the condition~~. The 24 hour Completion Time is reasonable based on the low probability of a postulated accident occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the safeguard buildings or fuel building boundary.

C.1 and C.2

In MODE 1, 2, 3, or 4, when Required Action A.1 or B.1 cannot be completed within the associated Completion Time, or when both SBVS Accident Exhaust Filtration trains are inoperable for reasons other than an inoperable safeguard building or fuel building boundary (i.e., Condition B), 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 MODE 3 within 6 hours and in MODE 5 within 36 hours. The 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.

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Storage Pool Boron Concentration

~~---Reviewer's Note---~~
~~The design of the spent fuel storage racks is to be provided by the COLA applicant. The required boron concentration will be provided as a part of the spent fuel rack design.~~
~~---~~

BASES

BACKGROUND	The water in the spent fuel storage pool normally contains soluble boron, which would result in large subcriticality margins under actual operating conditions. For storage of fuel in the spent fuel storage racks, the design basis for preventing criticality outside the reactor is that there is a 95 percent probability at a 95 percent confidence level, without soluble boron, that the effective multiplication factor (k_{eff}) of the fuel assembly array will be less than 0.995, including uncertainties and tolerances. The NRC guidelines specify a limiting k_{eff} of 1.0 for normal storage in the absence of soluble boron. Hence, the design is based on the use of unborated water, which maintains a subcritical condition for the allowed loading patterns. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single independent accident need be considered at one time. For example, the only accident scenario that has a potential for more than negligible positive reactivity effect is an inadvertent misplacement of a new fuel assembly. This accident has the potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation with unborated water and no movement of assemblies may, therefore, be achieved by controlling the location of each assembly in accordance with LCO 3.7.16, “Spent Fuel Pool Storage.” Prior to movement of an assembly, it is necessary to perform SR 3.7.16.1.~~As described in the following LCO 3.7.16, "Spent Fuel Storage," fuel assemblies are stored in the spent fuel racks without restriction. Although the water in the spent fuel pool is normally borated to ≥ [1291] ppm with boric acid enriched to ≥ 37% B^{10}, the criteria that limit storage of a fuel assembly to specific locations are conservatively developed without taking credit for boron.~~

APPLICABLE Although credit for the soluble boron normally present in the spent fuel ~~----~~
~~------------------------------REVIEWER'S NOTE-------------------------------~~
SAFETY pool water is permitted under abnormal or accident conditions, most ~~The design of the spent fuel storage racks is the responsibility of the COL~~
SES abnormal or accident conditions will not result in exceeding the limiting ~~applicant. A COL applicant that references the U.S. EPR design certification will demonstrate that the design satisfies the criticality analysis requirements for the spent fuel storage racks.~~
~~--~~

reactivity even in the absence of soluble boron. The effects on reactivity of credible abnormal and accident conditions due to temperature increase, boiling, assembly dropped on top of a rack, lateral rack module movement and misplacement of a spent fuel assembly have been analyzed. The spent fuel pool k_{eff} storage limit of 0.95 is maintained during these events by a minimum boron concentration of 500 ppm with boric acid enriched to ≥ 37% B^{10} established by criticality analysis (Ref. 2). Compliance with the LCO minimum boron concentration limit of 500 ppm with boric acid enriched to ≥ 37% B^{10} ensures that the credited concentration is always available.

The concentration and enrichment of dissolved boron in the spent fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO

The spent fuel storage pool boron concentration is required to be ≥ ~~[1291]~~500 ppm boron enriched to ≥ 37% B^{10}. The specified concentration and enrichment of dissolved boron in the spent fuel pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 2~~1~~. This concentration of dissolved boron is the minimum required concentration and enrichment for fuel assembly storage and movement within the spent fuel pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool, until a complete spent fuel pool verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

A.1, A.2.1, and A.2.2

When the concentration or enrichment of boron in the spent fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration or enrichment of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the spent fuel storage pool fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration and enrichment of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

~~ACTIONS~~

~~A.1, A.2.1, and A.2.2~~

~~When the concentration or enrichment of boron in the spent fuel pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration or enrichment of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the fuel storage pool fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration and enrichment of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.~~

~~The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.~~

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR and SR 3.7.15~~3~~.2 are met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

SR 3.7.15.2

Verification every 24 months that the B^{10} enrichment is within limit ensures that the B^{10} concentration assumed in the accident analyses is available. Since the boron in the spent fuel pool is not exposed to a significant neutron flux, 24 months is considered conservative.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2).

2. ~~[Holtec Topical Report]~~UN-TR-08-001, “Spent and New Fuel Storage Analyses for U.S. EPR Topical Report,” March 2008.~~ANSI/ANS 8.1-1998 "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."~~

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Storage

~~---Reviewer's Note---~~
~~The design of the spent fuel storage racks is to be provided by the COLA applicant. The required spent fuel storage configuration will be provided as a part of the spent fuel rack design.~~
~~---~~

BASES

BACKGROUND

The high density spent fuel storage racks are divided into two separate and distinct regions as shown in Figure 4.3-1. Region 1, with a maximum of 360 storage locations, is designed to accommodate new fuel assemblies with a maximum enrichment of 5.0 weight percent U-235, or spent fuel assemblies regardless of the combination of initial enrichment and burnup. Region 2, with a maximum of 1000 storage locations, is designed to accommodate spent fuel assemblies in all locations which comply with the combination of initial enrichment and burnup limits specified in Figure 3.7.16-1, Fuel Assembly Burnup Requirements for Region 2.

The water in the spent fuel storage pool normally contains soluble boron, which would result in large subcriticality margins under actual operating conditions. For storage of fuel in the spent fuel racks, the design basis for preventing criticality outside the reactor is that there is a 95 percent probability at a 95 percent confidence level, without soluble boron, that the effective multiplication faction (k_{eff}) of the fuel assembly array will be less than 0.995, including uncertainties and tolerances. The NRC guidelines specify a limiting k_{eff} of 1.0 for normal storage in the absence of soluble boron. Hence, the design is based on the use of unborated water, which maintains a subcritical condition for the allowed loading patterns.

The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal and accident conditions, since only a single independent accident need be considered at one time. For example, the only accident scenario that has the potential for more than negligible positive reactivity effect is an inadvertent misplacement of a new fuel assembly. This accident has is a potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. To mitigate these postulated criticality related accidents, enriched boron is dissolved in the pool water. Safe operation with unborated water and no movement of assemblies may, therefore, be achieved by controlling the combination of initial enrichment and burnup of the stored fuel in accordance with the accompanying LCO. Prior to movement of an assembly, it is necessary to perform SR 3.7.16.1.

BASES~~The spent fuel storage facility as described in Ref. 1 has a capacity of at least [1020] fuel assemblies. The spent fuel storage racks are designed to accommodate fuel with a maximum nominal enrichment of 5.0 wt% U-235 (maximum tolerance of +/- 0.05 wt%). The spent fuel storage racks are designed for unrestricted spent fuel assembly storage.~~

APPLICABLE SAFETY ANALYSES

The hypothetical accidents can only take place during or as a result of the movement of an assembly (Refs. 2 and 3). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.15, "Spent Fuel Pool Boron Concentration") prevents criticality. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the auspices of the accompanying LCO.~~The criticality analysis shows that the fuel remains subcritical under all credible abnormal conditions.~~

~~-----REVIEWER'S NOTE-----~~
~~The design of the spent fuel storage racks is the responsibility of the COL applicant. A COL applicant that references the U.S. EPR design certification will demonstrate that the design satisfies the criticality analysis requirements for the spent fuel storage racks.~~
~~----------~~

~~The design shows acceptable prevention of an increase in effective multiplication factor (k-effective) beyond safe limits based on the guidelines in Ref. 2.~~

The configuration of fuel assemblies in the spent fuel storage pool satisfies
——Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within Region 2 of the spent fuel pool in the accompanying LCO, ensure the k_{eff} of the spent fuel storage pool will always remain < 0.995, assuming the pool to be flooded with unborated water and < 0.95, with a boron concentration of greater than 500 ppm and boron enrichment ≥ 37%.

Storage of spent fuel is permitted in all Region 2 locations provided that the spent fuel meets the combination of initial enrichment and burnup requirements shown in Figure 3.7.16-1, Fuel Assembly Burnup Requirements for Region 2.~~Unrestricted storage of fuel assemblies within the spent fuel pool is allowed provided that the maximum nominal U-235 enrichment is equal to or less than 5.00 weight percent. This ensures the k-effective of the spent fuel pool will always remain less than 0.95 assuming the pool is flooded with [unborated water].~~

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region 2 of the spent fuel storage pool.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region 2 of the spent fuel storage pool.

ACTIONS

A.1

When the requirements of the LCO are not met, action must be immediately initiated to move the non-complying fuel assembly to an acceptable storage location (i.e., Region 1).

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

~~ACTIONS~~ ~~A.1~~

~~When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with the LCO, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance.~~

~~Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.~~

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by administrative means that ~~the fuel assembly is in accordance with the configurations specified in the accompanying LCO~~ initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.16-1 prior to storing the fuel assembly in Region 2.

REFERENCES

1. Double contingency principle ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2).~~FSAR Section 9.1.2~~

2. ~~2.~~ UN-TR-08-001, “Spent and New Fuel Storage Analyses for U.S. EPR Topical Report,” March 2008.~~[Holtec Topical Report]10CFR 50.68, "Criticality Accident Requirements."~~

3. U.S. EPR FSAR Section 15.0.3.10, “Fuel Handling Accident.”