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Your ref: Project Number 740  
Our ref: DCP/NRC1807

December 15, 2006

Subject: AP1000 COL Standard Technical Report Submittal

In support of Combined License application pre-application activities, Westinghouse is submitting Revision 0 of AP1000 Standard Combined License Technical Report Number 74A. This report completes and documents, on a generic basis, activities required for COL Information Item 16.1-1 in the AP1000 Design Control Document. Changes to the Design Control Document identified in Technical Report Number 74A are intended to be incorporated into FSARs referencing the AP1000 design certification or incorporated into the design certification using supplemental rulemaking if Part 52 is revised to permit revision of the design certification. This report is submitted as part of the NuStart Bellefonte COL Project (NRC Project Number 740). The information included in this report is generic and is expected to apply to all COL applications referencing the AP1000 Design Certification.

The purpose for submittal of this report was explained in a March 8, 2006 letter from NuStart to the U.S. Nuclear Regulatory Commission.

Pursuant to 10 CFR 50.30(b), APP-GW-GLR-064, Revision 0, "AP1000 Generic Technical Specifications Completion," Technical Report Number 74A, is submitted as Enclosure 1 under the attached Oath of Affirmation.

Attachment 2 provides a list of open items from Technical Report 74A. A schedule for addressing these open items will be provided to the NRC by January 31, 2007.

An additional Technical Report is planned to identify any technical specification changes necessary to support AP1000 plant design changes. The technical report information schedule and content will be defined in the January DCWG Technical Report status letter.

It is expected that when the NRC review of Technical Report Number 74A is complete and the remaining open items addressed, COL Information Item 16.1-1 will be considered complete for COL applicants referencing the AP1000 Design Certification.

Questions or requests for additional information related to the content and preparation of this report should be directed to Westinghouse. Please send copies of such questions or requests to the prospective applicants for combined licenses referencing the AP1000 Design Certification. A representative for each applicant is included on the cc: list of this letter.

Very truly yours,



A. Sterdis, Manager  
Licensing and Customer Interface  
Regulatory Affairs and Standardization

/Attachment

1. "Oath of Affirmation," dated December 15, 2006
2. List of Open Items from Technical Report Number 74A

/Enclosure

1. APP-GW-GLR-064, Revision 0, "AP1000 Generic Technical Specifications Completion," Technical Report Number 74A, dated December 2006.

cc:	S. Bloom	- U.S. NRC	1E	1A
	S. Coffin	- U.S. NRC	1E	1A
	G. Curtis	- TVA	1E	1A
	P. Grendys	- Westinghouse	1E	1A
	P. Hastings	- Duke Power	1E	1A
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	A. Monroe	- SCANA	1E	1A
	M. Moran	- Florida Power & Light	1E	1A
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	E. Schmiech	- Westinghouse	1E	1A
	G. Zinke	- NuStart/Entergy	1E	1A

ATTACHMENT 1

“Oath of Affirmation”

ATTACHMENT 1

UNITED STATES OF AMERICA  
NUCLEAR REGULATORY COMMISSION

In the Matter of: )  
NuStart Bellefonte COL Project )  
NRC Project Number 740 )

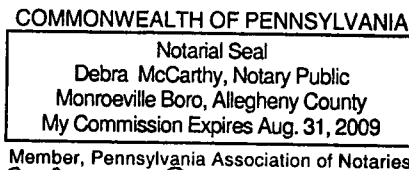
APPLICATION FOR REVIEW OF  
"AP1000 GENERAL COMBINED LICENSE INFORMATION"  
FOR COL APPLICATION PRE-APPLICATION REVIEW

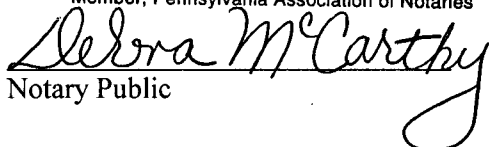
W. E. Cummins, being duly sworn, states that he is Vice President, Regulatory Affairs & Standardization, for Westinghouse Electric Company; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission this document; that all statements made and matters set forth therein are true and correct to the best of his knowledge, information and belief.



W. E. Cummins  
Vice President  
Regulatory Affairs & Standardization

Subscribed and sworn to  
before me this 15th day  
of December 2006.



  
Notary Public

ATTACHMENT 2

“List of Open Items from Technical Report 74A”

**List of Open Items from TR74A**

<b>TS Item</b>	<b>Comments</b>
<b>Design Technical Actions</b>	
TSs 3.3.1 and 3.3.2 Digital I&C SRs and Completion Times	Technical work is complete but AP1000-specific design documents not verified
TS 3.4.6 Overpressure Report	Technical work to be completed during NRC review
TS 3.4.14 RCS vent area	Awaiting vendor confirmation of valve performance to confirm need for pipe ID design change.
TSs 3.6.8 and 3.9.5 Equipment hatch bolts	Preliminary design calculation completed and needs verified
TS 3.6.9 Post-accident water volume, accident [B], TSP mass, and TSP sample mass	Preliminary design calculation completed and needs verified
TSs 3.9.5 and 3.9.6 VFS differential pressure	TS limit specified in system design documents Will verify in design performance calculation
<b>Plant Information Actions</b>	
TS 4.1 Site-specific admin note	Information to be provided by COL if generic agreement is not achieved
TS 4.1.1 Site and exclusion zone information	Information to be provided by COL if generic agreement is not achieved
TS 4.1.2 LPZ information	Information to be provided by COL if generic agreement is not achieved
TSs 5.1 and 5.2 Admin Controls Responsibility and Organization	Information to be provided by COL if generic agreement is not achieved
TS 5.3 Admin Controls Unit Staff Qualifications	Information to be provided by COL if generic agreement is not achieved
TSs 5.6.1 and 5.6.2 COL date for submitting initial Occup Rad Expose Report and RA BTP version	Information to be provided by COL if generic agreement is not achieved

ENCLOSURE 1

APP-GW-GLR-064, Revision 0

AP1000 Generic Technical Specifications Completion

Technical Report Number 74A

# AP1000 DOCUMENT COVER SHEET

TDC: \_\_\_\_\_ Permanent File: \_\_\_\_\_ APY: \_\_\_\_\_

RFS#: \_\_\_\_\_ RFS ITEM #: \_\_\_\_\_

AP1000 DOCUMENT NO. APP-GW-GLR-064	REVISION NO. 0	Page 1 of 149	ASSIGNED TO W-A. Sterdis
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ALTERNATE DOCUMENT NUMBER: TR-74A

WORK BREAKDOWN #:

ORIGINATING ORGANIZATION: Westinghouse Electric Company

TITLE: AP1000 Generic Technical Specifications Completion

ATTACHMENTS:	DCP #/REV. INCORPORATED IN THIS DOCUMENT REVISION:
CALCULATION/ANALYSIS REFERENCE:	

ELECTRONIC FILENAME	ELECTRONIC FILE FORMAT	ELECTRONIC FILE DESCRIPTION
APP-GW-GLR-064 Rev 0.doc	Microsoft Word	

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LEGAL REVIEW	SIGNATURE/DATE
PATENT REVIEW	SIGNATURE/DATE

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ORIGINATOR C.E. Morgan	SIGNATURE/DATE <i>C.E. Morgan</i> for 12/15/06	
REVIEWERS C.S. Brockhoff	SIGNATURE/DATE <i>C.S. Brockhoff</i> 12/15/06	
VERIFIER F.B. Baskerville	SIGNATURE/DATE <i>F.B. Baskerville</i> 12/15/06	VERIFICATION METHOD Confirmed all changes
AP1000 RESPONSIBLE MANAGER J.W. Winters	SIGNATURE <i>J.W. Winters</i>	APPROVAL DATE 12/15/06

\* Approval of the responsible manager signifies that document is complete, all required reviews are complete, electronic file is attached and document is released for use.



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APP-GW-GLR-064  
Revision 0

November 2006

# **AP1000 Standard Combined License Technical Report TR-74A**

## **AP1000 Generic Technical Specifications Completion**

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# 1 INTRODUCTION

The purpose of this report is to provide updated technical information for the bracketed items in technical specifications in the AP1000 DCD, as discussed in the Combined License Information in DCD 16.1.1 and in the AP1000 Final Safety Evaluation Report (FSER).

## 1.1 COL INFORMATION ITEM 16.1-1

### “DCD 16.1.1 Combined License Information”

This set of technical specifications is intended to be used as a guide in the development of the plant-specific technical specifications. Combined License applicants referencing the AP1000 will replace preliminary information provided in brackets [ ] with final plant specific values.”

### FSER 16.2 (NUREG-1793)

### “Technical Specifications (TS)”

In some instances, detailed design information, equipment selection, allowable values, or other information are needed to establish the information to be included in the TS. Locations for the addition of this information are signified by brackets to indicate that the combined license (COL) applicant must provide plant-specific values or alternative text. This is COL Action Item 16.2-1.”

## 1.2 GENERAL

The purpose of this report is to provide the following:

- Identify changes to the generic technical specifications contained in the DCD to incorporate final design information. This will ultimately reduce the number of Information items required by COL Information Item 16.1-1. The changes will include the justifications or reference to supporting justifications. The intent is to accomplish this by submitting revised pages that would be used to update the generic technical specification pages contained in the DCD.

Appendix A contains markups of the proposed changes to address COL Information Item 16.1-1.

### **Exemption Request for Appendix A**

Some of the changes in Appendix A also included information that is located outside of the brackets. The information provided outside the brackets is directly related to the completion of the bracketed information item and is supported by the same source document provided to complete the bracketed information item.

This report supports an exemption for these items consistent with the requirements of Part 52, Appendix D, §VIII.C.4. The proposed changes are acceptable under 10 CFR 50.12(a)(1) in that the proposed changes will not present an undue risk to the public health and safety, and are consistent with the common defense and security. The proposed changes are needed per 10 CFR 50.12(a)(2)(ii) in that the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

The specific items being revised that are located outside of brackets include the following:

1. Bases for TS 3.2.1 ( $F_Q$  Methodology), page B 3.2.1-3

This bracketed peaking factor value may be supplemented by an appropriate peaking factor that will be specified in the AP1000 COLR and, therefore, the bracketed value by itself, without the modifying text revisions, is incorrect for the AP1000 design.

2. Allowable Values in Tables 3.3.1-1 and 3.3.2-1

These allowable values are integral to providing the value for the bracketed trip setpoints using the AP1000 setpoint methodology, as reflected in the Reviewer Note for each table. The Reviewer Notes specify that the trip setpoint values would be updated and that allowable values would be calculated and provided in the appropriate table columns. Therefore, the allowable values are provided as part of the resolution of these bracketed items.

3. Units for Table 3.3.1, Items 13 and 16 (SG narrow range water level and intermediate range neutron flux instruments)

The units for these two items changed as a result of continuing plant design that established the actual indication values provided by each instrumentation channel. Therefore, the generic NUREG-1431, Rev. 2 information that formed the basis for the original DCD units is incorrect for the current AP1000 design, by solely making changing within the bracketed information.

4. Note 1 for Table 3.3.1 (determination of the overtemperature delta-T setpoint for Item 6)

The values in brackets that were originally based on generic NUREG-1431, Rev. 2 information are inadequate to completely characterize the setpoint and allowable value for AP1000 by solely making changing within the bracketed information. Therefore, additional text changes are required to correctly describe these values.

5. Note 2 of TS 3.4.4 and 3.4.8

Note 2 was added to each of these TSs to be consistent with this note which was approved by the staff in TS 3.4.14. These changes do not result in a technical changes to technical specifications since this limitation was approved for the current technical specifications. In correcting the bracketed information for the note(s) related to starting RCPs, it was decided that it would be appropriate to add the note from in TS 3.4.14 to TSs 3.4.4 and 3.4.8 for completeness to minimize the potential for operational errors.

6. Bases for TS 3.1.1 (Shutdown Margin)

The Bases discussion includes an example calculating the shutdown margin that was based on generic NUREG-1431, Rev. 2 information. The generic information was inadequate to completely characterize the calculation for the AP1000 example by solely making changing within the generic bracketed information.

The markups in Appendix A are annotated with the following:

- 1 The preliminary information contained in the brackets has been replaced with the final information and is based on system design specifications, approved engineering calculation notes and/or verified analysis input assumptions.
- 2 The preliminary information has been replaced with the final information and is based on WCAP-15025-P-A.
- 3 Not used.
- 4 The preliminary information has been replaced with the final information and is based on engineering judgment.
- 5 The preliminary information has been replaced with the final information and is based on the design capability of the system.
- 6 The preliminary information has been replaced with the final information and is based on historical relationships between the no load operating temperature (557°F), the minimum temperature for criticality (551°F), and the limit for MODE 2 physics testing (541°F).
- 7 The preliminary information is still under evaluation and will be provided during the NRC review.
- 8 The preliminary information has been replaced with the updated information and is based on WCAP-16361-P.
- 9 Not used.
- 10 The preliminary information has been replaced with the updated information and is based on WCAP-16361-P indication uncertainty for pressurizer level indication.

- 11 The preliminary information will be retained in brackets until a plant specific value can be determined using as built information.
- 12 The preliminary information has been replaced with final information and is based on the response to NRC RAI 440.106.
- 13 The preliminary information is no longer applicable due to a planned design change. Replacement pages will be provided later in a separate report to address technical specifications impacted by design changes.
- 14 The preliminary information will be retained in brackets until a plant specific value can be determined using as built information. The bracketed information was changed to be consistent with TS 3.8.7 Action B.2.
- 15 Not used.
- 16 Not used.
- 17 The site specific information will be provided by the specific COL Applicant in the Plant Specific Technical Specifications for that applicant. Any reviewer's notes should be removed at that time
- 18 The preliminary information has been replaced with final information and is based on the the nominal RCS pressure design of AP1000 and the requirements for test pressures identified in ASME OM Code ISTC-3630(b).

## 1.3 REGULATORY IMPACT

### A. FSER IMPACT

The AP1000 Generic Technical Specifications are discussed in Chapter 16 of the NRC Final Safety Evaluation Report (FSER). The write-up describing the development of the technical specification and the NRC review are not impacted by the changes to the technical specifications identified in Appendix A. The changes in APP-GW-GLR-064 Appendix A include completion of the majority of the preliminary information identified in brackets in the technical specifications. The information not provided at this time is generally information that cannot be provided until as-built information is available or is related to plant specific site location and organization. The changes proposed in Appendix A will not impact the FSER conclusion that the proposed AP1000 technical specifications are consistent with the regulatory guidance contained in the standard technical specifications.

### B. EVALUATION OF DEPARTURE FROM TIER 2 INFORMATION

(Check correct response and provide justification for that determination under each response)

10 CFR Part 52, Appendix D, Section VIII.B.5.a. provides that an applicant for a combined licensee who references the AP1000 design certification may depart from Tier 2 information, without prior NRC approval, if it does not require a license amendment under paragraph B.5.b. The questions below address the criteria of B.5.b.

1. Does the proposed departure result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the plant-specific DCD?

The technical specification changes described in APP-GW-GLR-064 will not increase the frequency of occurrence of an accident because there is no significant increase in the probability of failure of the initiation, performance, and pressure boundary integrity design functions of the reactor coolant system and engineered safety features.

2. Does the proposed departure result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety and previously evaluated in the plant-specific DCD?

The technical specification changes described in APP-GW-GLR-064 do not affect the initiation, performance, and pressure boundary integrity design functions of the reactor coolant system and engineered safety features, therefore there is no increase in the probability of malfunctions of these safety functions.

3. Does the proposed departure result in more than a minimal increase in the consequences of an accident previously evaluated in the plant-specific DCD?

The technical specification changes described in APP-GW-GLR-064 have no effect on the initiation, performance, and pressure boundary integrity design functions of the reactor coolant system and engineered safety features and will not increase the release of radioactivity during a postulated accident. Therefore, there is no impact on the consequences of an accident.

4. Does the proposed departure result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the plant-specific DCD?

The technical specification changes described in APP-GW-GLR-064 will not impact the initiation, performance, and pressure boundary integrity design functions of the reactor coolant system and engineered safety features and will not increase the release of radioactivity during a postulated malfunction of an SSC. Therefore, there is no increase the consequences of a malfunction of an SSC important to safety.

5. Does the proposed departure create a possibility for an accident of a different type than any evaluated previously in the plant-specific DCD?

No, the technical specification changes described in APP-GW-GLR-064 will not impact the response of the reactor coolant system or engineered safety features to postulated accident conditions. The changes also do not introduce any additional failure modes. Therefore, these changes will not result in an accident of a type different than what has already been evaluated in the DCD.

6. Does the proposed departure create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously in the plant-specific

No, the technical specification changes described in APP-GW-GLR-064 will not result in any impact to the safety functions of the initiation, performance, and pressure boundary integrity design functions of the reactor coolant system and engineered safety features, and therefore there it will not impact a malfunction of an SSC to cause a different result than what has been evaluated previously.

7. Does the proposed departure result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered

No, the technical specification changes described in APP-GW-GLR-064 will not result in any impact to initiation, performance, and pressure boundary integrity design functions of the reactor coolant system and engineered safety features and thus will not result in a design basis limit for a fission product barrier being exceeded.

8. Does the proposed departure result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses

No, the technical specification changes described in APP-GW-GLR-064 will not alter the methodology used in verifying initiation, performance, and pressure boundary integrity design functions of the reactor coolant system and engineered safety features, or in performing the safety analyses.

The answers to the evaluation questions above are “NO” and the proposed departure from Tier 2 does not require prior NRC review to be included in plant specific FSARs as provided in 10 CFR Part 52, Appendix D, Section VIII. B.5.b

### **C. IMPACT ON RESOLUTION OF A SEVERE ACCIDENT ISSUE**

10 CFR Part 52, Appendix D, Section VIII. B.5.a. provides that an applicant for a combined licensee who references the AP1000 design certification may depart from Tier 2 information, without prior NRC approval, if it does not require a license amendment under paragraph B.5.c. The questions below address the criteria of B.5.c.

1. Does the proposed activity result in an impact on features that mitigate severe accidents.



No, the technical specification changes described in APP-GW-GLR-064 will not have an impact on the initiation, performance, and pressure boundary integrity design functions of the reactor coolant system and engineered safety features or any features that mitigate severe accidents.

The answers to the evaluation questions above are "NO" or are not applicable and the proposed departure from Tier 2 does not require prior NRC review to be included in plant specific FSARs as provided in 10 CFR Part 52, Appendix D, Section VIII. B.5.c

#### **D. SECURITY ASSESSMENT**

Does the proposed change have an adverse impact on the security assessment of the AP1000.

No, the technical specification completion described in APP-GW-GLR-064 will not alter barriers or alarms that control access to protected areas of the plant. The changes to the technical specifications will not alter requirements for security personnel.

#### **DCD Mark-ups**

Appendix A contains the markups of the Specifications and Bases associated with the information item discussed below.

Revise COL Information Item 16.1-1 in Subsection 16.1.1 as follows:

##### Combined License Information

This set of technical specifications is intended to be used as a guide in the development of the plant-specific technical specifications. The preliminary information originally provided in brackets [ ] has been revised with updated information (Reference 1, APP-GW-GLR-064). Combined License applicants referencing the AP1000 will be required to provide the final information for the remaining brackets [ ] with final plant specific information.

Add the following reference to DCD Section 16.1

#### 16.1.2 References

1. APP-GW-GLR-064, Revision 0, "AP1000 Generic Technical Specifications Completion," November 2006.

# APPENDIX A

## 2.0 SAFETY LIMITS (SLs)

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### 2.1 SLs

#### 2.1.1 Reactor Core SLs

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

- ② 2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained  $\geq$  {1.14 for the WRB-2M DNB correlations}.
- ① 2.1.1.2 The peak fuel centerline temperature shall be maintained  $<$  {5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup}.

#### 2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5 the RCS pressure shall be maintained  $\leq$  2733.5 psig.

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### 2.2 SL Violations

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

2.2.2 If SL 2.1.2 is violated:

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

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

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.3  11	<p>Verify rod drop time of each rod, from the fully withdrawn position, is <math>\leq [2.47]</math> seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with:</p> <ul style="list-style-type: none"> <li>a. <math>T_{avg} \geq 500^{\circ}\text{F}</math>, and</li> <li>b. All reactor coolant pumps operating.</li> </ul>	<p>Prior to reactor criticality after each removal of the reactor head</p>

ACTIONS (continued)

E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours
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SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.7.1 (5) Verify each DRPI agrees within 12 steps of the group demand position for the [full indicated range] of rod travel.	Prior to criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of:

- LCO 3.1.3 "Moderator Temperature Coefficient,"
- LCO 3.1.4 "Rod Group Alignment Limits,"
- LCO 3.1.5 "Shutdown Bank Insertion Limit,"
- LCO 3.1.6 "Control Bank Insertion Limits," and
- LCO 3.4.2 "RCS Minimum Temperature for Criticality"

may be suspended, and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, and 16.c, may be reduced to 3 provided:

6

- a. RCS lowest loop average temperature is  $\geq$  ~~535~~541°F,
- b. SDM is within the limits specified in the COLR, and
- c. THERMAL POWER is < 5% RTP.

APPLICABILITY: During PHYSICS TESTS initiated in MODE 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes
	<u>AND</u>	
	A.2 Suspend PHYSICS TESTS exceptions.	1 hour
B. THERMAL POWER not within limit.	B.1 Open reactor trip breakers.	Immediately
C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and Associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per SR 3.3.1.7.	Prior to initiation of PHYSICS TESTS
SR 3.1.8.2	Verify the RCS lowest loop average temperature is $\geq$ <del>535</del> 541°F.	30 minutes
SR 3.1.8.3	Verify THERMAL POWER is < 5% RTP.	30 minutes
SR 3.1.8.4	Verify SDM is within the limits specified in the COLR.	24 hours

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.2.1.2</p> <p style="text-align: center;">----- - NOTE - -----</p> <p>If <math>F_Q^W(Z)</math> measurements indicate</p> <p style="margin-left: 40px;">(1) maximum over <math>z\{F_Q^C(Z)\}</math></p> <p>has increased since the previous evaluation of <math>F_Q^C(Z)</math>:</p> <p>a. Increase <math>F_Q^W(Z)</math> by the <b>greater of a factor of 1.02 or by an appropriate factor specified in the COLR</b> and reverify <math>F_Q^W(Z)</math> is within limits; or</p> <p>b. Repeat SR 3.2.1.2 once per 7 EFPD until two successive flux maps indicate</p> <p style="margin-left: 40px;">maximum over <math>z\{F_Q^C(Z)\}</math></p> <p style="margin-left: 40px;">has not increased.</p> <p style="text-align: center;">-----</p> <p>Verify <math>F_Q^W(Z)</math> within limits.</p>	<p>Once after each refueling prior to THERMAL POWER exceeding 75% RTP</p> <p><u>AND</u></p> <p>Once within 12 hours after achieving equilibrium conditions after exceeding, by <math>\geq 10\%</math> RTP, the THERMAL POWER at which <math>F_Q^W(Z)</math> was last verified</p> <p><u>AND</u></p> <p>31 EFPD thereafter</p>



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. One or two Power Range Neutron Flux – High channels inoperable.</p>	<p>D.1.1 Reduce THERMAL POWER to <math>\leq 75\%</math> RTP.</p>	<p>12 hours</p>
	<p><u>AND</u></p>	
	<p>D.1.2 Place one inoperable channel in bypass or trip.</p>	<p>[6] hours (7)</p>
	<p><u>AND</u></p>	
	<p>D.1.3 With two inoperable channels, place one channel in bypass and one channel in trip.</p>	<p>[6] hours (7)</p>
	<p><u>OR</u></p>	
<p>E. One or two channels inoperable.</p>	<p>D.2.1 Place inoperable channel(s) in bypass.</p>	<p>[6] hours (7)</p>
	<p><u>AND</u></p>	
	<p>-----  <b>- NOTE -</b>                      Only required to be performed when OPDMS is inoperable and the Power Range Neutron Flux input to QPTR is inoperable.                      -----</p>	
	<p>D.2.2 Perform SR 3.2.4.2 (QPTR verification).</p>	<p>Once per 12 hours</p>
<p><u>OR</u></p>		
<p>D.3 Be in MODE 3.</p>	<p>12 hours</p>	
<p>E. One or two channels inoperable.</p>	<p>E.1.1 Place one inoperable channel in bypass or trip.</p>	<p>[6] hours (7)</p>
	<p><u>AND</u></p>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	<p>E.1.2 With two channels inoperable, place one channel in bypass and one channel in trip.</p> <p><u>OR</u></p> <p>E.2 Be in MODE 3.</p>	<p>[6] hours (7)</p> <p>12 hours</p>
<p>F. THERMAL POWER between P-6 and P-10, one or two Intermediate Range Neutron Flux channels inoperable.</p>	<p>F.1.1 Place one inoperable channel in bypass or trip.</p> <p><u>AND</u></p> <p>F.1.2 With two channels inoperable, place one channel in bypass and one channel in trip.</p> <p><u>OR</u></p> <p>F.2 Reduce THERMAL POWER to &lt; P-6.</p> <p><u>OR</u></p> <p>F.3 Increase THERMAL POWER to &gt; P-10.</p>	<p>[2] hours (7)</p> <p>[2] hours (7)</p> <p>2 hours</p> <p>2 hours</p>
<p>G. THERMAL POWER between P-6 and P-10, three Intermediate Range Neutron Flux channels inoperable.</p>	<p>G.1 Suspend operations involving positive reactivity additions.</p> <p><u>AND</u></p> <p>G.2 Reduce THERMAL POWER to &lt; P-6.</p>	<p>Immediately</p> <p>2 hours</p>
<p>H. THERMAL POWER &lt; P-6, one or two Intermediate Range Neutron Flux channels inoperable.</p>	<p>H.1 Restore three of four channels to OPERABLE status.</p>	<p>Prior to increasing THERMAL POWER to &gt; P-6</p>

ACTIONS (continued)

CONDITION		REQUIRED ACTION	COMPLETION TIME	
I.	One or two Source Range Neutron Flux channels inoperable.	I.1 Suspend operations involving positive reactivity additions.	Immediately	
J.	Three Source Range Neutron Flux channels inoperable.	J.1 Open RTBs.	Immediately	
K.	One or two channels inoperable.	K.1.1 Place one inoperable channel in bypass or trip.	[6] hours	7
		<u>AND</u>		
		K.1.2 With two channels inoperable, place one channel in bypass and one channel in trip.	[6] hours	7
L.	One or two channels inoperable.	<u>OR</u>		
		K.2 Reduce THERMAL POWER to < P-10.	12 hours	
		L.1.1 Place one inoperable channel in bypass or trip.	[6] hours	7
L.	One or two channels inoperable.	<u>AND</u>		
		L.1.2 With two channels inoperable, place one channel in bypass and one channel in trip.	[6] hours	7
		<u>OR</u>		
M.	One or two channels/divisions inoperable.	L.2 Reduce THERMAL POWER to < P-8.	10 hours	
		M.1 Restore three of four channels/divisions to OPERABLE status.	6 hours	
M.	One or two channels/divisions inoperable.	<u>OR</u>		
		M.2 Be in MODE 3.	12 hours	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>N. One or two interlock channels inoperable.</p>	<p>N.1 Verify the interlocks are in required state for existing plant conditions.</p>	<p>1 hour</p>
	<p><u>OR</u></p>	
	<p>N.2.1 Place the Functions associated with one inoperable interlock channel in bypass or trip.</p>	<p>[7] hours (7)</p>
	<p><u>AND</u></p>	
	<p>N.2.2 With two interlock channels inoperable, place the Functions associated with one inoperable interlock channel in bypass and with one inoperable interlock channel in trip.</p>	<p>[7] hours (7)</p>
<p><u>OR</u></p>		
<p>O. One or two interlock channels inoperable.</p>	<p>N.3 Be in MODE 3.</p>	<p>13 hours</p>
	<p><u>OR</u></p>	
	<p>O.1 Verify the interlocks are in required state for existing plant conditions.</p>	<p>1 hour</p>
<p><u>OR</u></p>		
<p>O.2.1 Place the Functions associated with one inoperable interlock channel in bypass.</p>	<p><u>AND</u></p>	<p>[7] hours (7)</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	<p>O.2.2 With two interlock channels inoperable, place the Functions associated with one inoperable interlock channel in bypass and with one inoperable interlock channel in trip.</p> <p><u>OR</u></p> <p>O.3 Be in MODE 2.</p>	<p>[7] hours <span style="border: 1px solid black; border-radius: 50%; padding: 2px 5px;">7</span></p> <p>13 hours</p>
<p>P. One division inoperable.</p>	<p>P.1 Open RTBs in inoperable division.</p> <p><u>OR</u></p> <p>P.2.1 Be in MODE 3, 4, or 5.</p> <p><u>AND</u></p> <p>P.2.2 Open RTBs.</p>	<p>8 hours</p> <p>14 hours</p> <p>14 hours</p>
<p>Q. Two divisions inoperable.</p>	<p>Q.1 Restore three of four divisions to OPERABLE status.</p> <p><u>OR</u></p> <p>Q.2.1 Be in MODE 3, 4, or 5.</p> <p><u>AND</u></p> <p>Q.2.2 Open RTBs.</p>	<p>1 hour</p> <p>7 hours</p> <p>7 hours</p>
<p>R. One or two channels/ divisions inoperable.</p>	<p>R.1 Restore three of four channels/divisions to OPERABLE status.</p> <p><u>OR</u></p> <p>R.2 Open RTBs.</p>	<p>48 hours</p> <p>49 hours</p>

ACTIONS (continued)

CONDITION		REQUIRED ACTION	COMPLETION TIME
S.	One or two Source Range Neutron Flux channel inoperable.	S.1 Restore three of four channels to OPERABLE status.	[48] hours (7)
		<u>OR</u>	
		S.2 Open RTBs.	[49] hours (7)
T.	Required Source Range Neutron Flux channel inoperable.	T.1 Suspend operations involving positive reactivity additions.	Immediately
		<u>AND</u>	
		T.2 Close unborated water source isolation valves.	1 hour
		<u>AND</u>	
		T.3 Perform SR 3.1.1.1.	1 hour
			<u>AND</u> Once per 12 hours thereafter

SURVEILLANCE REQUIREMENTS

- NOTE -

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

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours

SURVEILLANCE REQUIREMENTS (continued)


SURVEILLANCE		FREQUENCY
SR 3.3.1.4	<p>-----  <b>- NOTE -</b>                      Required to be met within 24 hours after reaching 50% RTP.                      -----</p> <p>Calibrate excore channels to agree with incore detector measurements.</p>	92 EFPD
SR 3.3.1.5	<p>-----  <b>- NOTE -</b>                      This Surveillance must be performed on both reactor trip breakers associated with a single division.                      -----</p> <p>Perform TADOT.</p>	92 days on a STAGGERED TEST BASIS
SR 3.3.1.6	<p>-----  <b>- NOTE -</b>                      Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3.                      -----</p> <p>Perform RTCOT.</p>	<div style="text-align: center;">  </div> <p>[92] days</p>

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.10	N/ANA	N/ANA
	3 <sup>(a)</sup> ,4 <sup>(a)</sup> ,5 <sup>(a)</sup>	2	C	SR 3.3.1.10	N/ANA	N/ANA
2. Power Range Neutron Flux						
a. High Setpoint	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.6 SR 3.3.1.9 SR 3.3.1.11	≤ 109.06% RTP	≤ [118]109% RTP
b. Low Setpoint	1 <sup>(b)</sup> ,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.11	≤ 25.06% RTP	≤ [35]25% RTP
3. Power Range Neutron Flux High Positive Rate	1,2	4	E	SR 3.3.1.6 SR 3.3.1.9 SR 3.3.1.11	≤ 5.06% RTP with time constant ≥ 2 sec	≤ [5.0] RTP with time constant ≥ [2] sec <sup>‡</sup>
4. Intermediate Range Neutron Flux	1 <sup>(b)</sup> ,2 <sup>(c)</sup>	4	F,G	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9	≤ 25.23% RTP*	≤ [25]% RTP*
	2 <sup>(d)</sup>	4	H	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9	≤ 25.23% RTP*	≤ [25]% RTP*
5. Source Range Neutron Flux High Setpoint	2 <sup>(d)</sup>	4	I,J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.11	≤ 1.01 E5 cps*	≤ [1.0 E5] cps*
	3 <sup>(a)</sup> ,4 <sup>(a)</sup> ,5 <sup>(a)</sup>	4	J,S	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.9 SR 3.3.1.11	≤ 1.01 E5 cps*	≤ [1.0 E5] cps*
	3 <sup>(e)</sup> ,4 <sup>(e)</sup> ,5 <sup>(e)</sup>	1	T	SR 3.3.1.1 SR 3.3.1.9	N/ANA	N/ANA

- (a) With Reactor Trip Breakers (RTBs) closed and Plant Control System capable of rod withdrawal.
- (b) Below the P-10 (Power Range Neutron Flux) interlocks.
- (c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.
- (d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
- (e) With RTBs open. In this condition, Source Range Function does not provide reactor trip but does provide indication.

[Reviewer Note: The values specified in brackets in the Trip Setpoint column are the Chapter 15 safety analysis values and are included for reviewer information only.

For the values specified in brackets followed by "\*" in the Trip Setpoint and Allowable Value columns, are typical values for the Function. No credit was assumed for these Functions (typically diverse trips/actuators) in the Chapter 15 safety analyses and no safety analysis value is available.

In all cases, the values specified for trip setpoints and allowable values in brackets must be replaced confirmed, following completion of the plant-specific setpoint study, with the actual Trip Setpoints. Upon selection of the plant specific instrumentation, the Trip Setpoints will be calculated in accordance with the setpoint methodology described in WCAP-16361-P-14606. (WCAP-14606 is an AP600 document that describes a methodology that is applicable to AP1000. AP1000 has some slight differences in instrument



spans as a result of the higher power level.) Allowable Values will be calculated in accordance with the setpoint methodology and specified in the Allowable Value column. The plant specific setpoint calculations will reflect the latest licensing analysis/design basis and may incorporate NRC accepted improvements in setpoint methodology.

Table 3.3.1-1 (page 2 of 5)  
Reactor Trip System Instrumentation

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FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6. Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11	Refer to Note 1 (Page 3.3.1-16)	Refer to Note 1 (Page 3.3.1-16)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11	Refer to Note 2 (Page 3.3.1-16)	Refer to Note 2 (Page 3.3.1-16)
8. Pressurizer Pressure	1 <sup>(f)</sup>	4	K	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11	≥ 1809.9 psig	≥ 1810.3 psig
a. Low Setpoint						
b. High Setpoint	1,2	4	E	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11	≤ 2420.7 psig	≤ 2420.3 psig
9. Pressurizer Water Level – High 3	1 <sup>(f)</sup>	4	K	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8	≤ 71.05%	≤ 71%±
10. Reactor Coolant Flow – Low						
a. Single Hot Leg	1 <sup>(g)</sup>	4 per hot leg	L	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11	≥ 89.96% <sup>(i)</sup>	≥ 90% <sup>(i)</sup>
b. Both Hot Legs	1 <sup>(h)</sup>	4 per hot leg	K	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11	≥ 89.96% <sup>(i)</sup>	≥ 90% <sup>(i)</sup>

(f) Above the P-10 (Power Range Neutron Flux) interlock.

(g) Above the P-8 (Power Range Neutron Flux) interlock.

(h) Above the P-10 (Power Range Neutron Flux) interlock and below the P-8 (Power Range Neutron Flux) interlock.

(i) Percent of thermal design flow 90% of loop specific indicated flow.

Table 3.3.1-1 (page 3 of 5)  
Reactor Trip System Instrumentation

8

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
11. Reactor Coolant Pump (RCP) Bearing Water Temperature – High						
a. Single Pump	1 <sup>(g)</sup>	4 per RCP	L	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8	≤ 230.4°F*	≥ <del>{320}</del> 230°F*
b. Two Pumps	1 <sup>(h)</sup>	4 per RCP	K	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8	≤ 230.4°F*	≥ <del>{320}</del> 230°F*
12. RCP Speed – Low						
	1 <sup>(f)</sup>	4	K	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8	≥ 90.9%	≥ <del>{90}</del> 91%
13. Steam Generator (SG) Narrow Range Water Level – Low						
	1,2	4 per SG	E	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11	≥ 20.95% span	≥ <del>{95,000}</del> lbm 21% span
14. Steam Generator (SG) Narrow Range Water Level – High 2						
	1,2 <sup>(k)</sup>	4 per SG	E	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11	≤ 82.05% span	≤ <del>{100}</del> 82% span
15. Safeguards Actuation Input from Engineered Safety Feature Actuation System						
a. Manual	1,2	2	B	SR 3.3.1.10	N/ANA	N/ANA
b. Automatic	1,2	4	M	SR 3.3.1.6	N/ANA	N/ANA

(f) Above the P-10 (Power Range Neutron Flux) interlock.

(g) Above the P-8 (Power Range Neutron Flux) interlock.

(h) Above the P-10 (Power Range Neutron Flux) interlock and below the P-8 (Power Range Neutron Flux) interlock.

(k) Above the P-11 (Pressurizer Pressure) interlock.

Table 3.3.1-1 (page 4 of 5)  
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
16. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2	4	N	SR 3.3.1.6 SR 3.3.1.9	≥ 9.91 E-6% RTP	≥ {1E-10}5% RTP-amps
b. Power Range Neutron Flux, P-8	1	4	O	SR 3.3.1.6 SR 3.3.1.9	≤ 48.06% RTP*	≤ {48}% RTP*
c. Power Range Neutron Flux, P-10	1,2	4	N	SR 3.3.1.6 SR 3.3.1.9	≥ 9.94% RTP ≤ 10.06% RTP	{10}% RTP
d. Pressurizer Pressure, P-11	1,2	4	N	SR 3.3.1.6 SR 3.3.1.9	≤ 1970.4 psig	≤ {1970} psig
17. Reactor Trip Breakers	1,2 3 <sup>(j)</sup> ,4 <sup>(j)</sup> ,5 <sup>(j)</sup>	4 divisions with 2 RTBs per division	P,Q	SR 3.3.1.5	N/ANA	N/ANA
18. Reactor Trip Breaker (RTB) Undervoltage and Shunt Trip Mechanisms	1,2 3 <sup>(j)</sup> ,4 <sup>(j)</sup> ,5 <sup>(j)</sup>	1 each per RTB mechanism	P,Q	SR 3.3.1.5	N/ANA	N/ANA
19. Automatic Trip Logic	1,2 3 <sup>(j)</sup> ,4 <sup>(j)</sup> ,5 <sup>(j)</sup>	4 divisions 4 divisions	M R	SR 3.3.1.6 SR 3.3.1.6	N/ANA N/ANA	N/ANA N/ANA
20. ADS Stages 1, 2, and 3 Actuation input from engineered safety feature actuation system						
a. Manual	1,2 3 <sup>(j)</sup> ,4 <sup>(j)</sup> ,5 <sup>(j)</sup>	2 switch sets 2 switch sets	B B	SR 3.3.1.10 SR 3.3.1.10	N/ANA N/ANA	N/ANA N/ANA
b. Automatic	1,2 3 <sup>(j)</sup> ,4 <sup>(j)</sup> ,5 <sup>(j)</sup>	4 4	M R	SR 3.3.1.6 SR 3.3.1.6	N/ANA N/ANA	N/ANA N/ANA
21. Core Makeup Tank Actuation input from engineered safety feature actuation system						
a. Manual	1,2 3 <sup>(j)</sup> ,4 <sup>(j)</sup> ,5 <sup>(j)</sup>	2 switch sets 2 switch sets	B B	SR 3.3.1.10 SR 3.3.1.10	N/ANA N/ANA	N/ANA N/ANA
b. Automatic	1,2 3 <sup>(j)</sup> ,4 <sup>(j)</sup> ,5 <sup>(j)</sup>	4 4	M R	SR 3.3.1.6 SR 3.3.1.6	N/ANA N/ANA	N/ANA N/ANA

(j) With Reactor Trip Breakers closed and Plant Control System capable of rod withdrawal.

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

8

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following nominal Trip Setpoint by more than ~~[(TBD)]%~~ of the following: For ΔT (0.10% ΔT span, for T<sub>avg</sub> (0.13% ΔT span), for pressure (0.04% ΔT span), for ΔI (0.09% ΔT span).

$$\Delta T \frac{(1 + \tau_4 s)}{(1 + \tau_5 s)} \leq \Delta T_0 \left\{ K_1 - K_2 \frac{(1 + \tau_1 s)}{(1 + \tau_2 s)} [T - T'] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.

ΔT<sub>0</sub> is the **loop specific** indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec<sup>-1</sup>.

T is the measured RCS average temperature, °F.

T' is the **loop specific** indicated T<sub>avg</sub> at RTP, ≤[\*]°F.

P is the measured pressurizer pressure, psig.

P' is the nominal RCS operating pressure, 2235 psig.

K<sub>1</sub> ≤ [\*]

K<sub>2</sub> ≥ [\*]/°F

K<sub>3</sub> ≥ [\*]/psig

τ<sub>1</sub> ≥ [\*] sec

τ<sub>2</sub> ≤ [\*] sec

τ<sub>4</sub> ≥ [\*] sec

τ<sub>5</sub> ≤ [\*] sec

f<sub>1</sub>(ΔI) = [\*] {[\*] + (q<sub>t</sub> - q<sub>b</sub>)}

when q<sub>t</sub> - q<sub>b</sub> ≤ -[\*]% RTP

0% of RTP

when -[\*]% RTP < q<sub>t</sub> - q<sub>b</sub> ≤ [\*]% RTP

-[\*] {(q<sub>t</sub> - q<sub>b</sub>) - [\*]}

when q<sub>t</sub> - q<sub>b</sub> > [\*]% RTP

Where q<sub>t</sub> and q<sub>b</sub> are percent RTP in the upper and lower halves of the core respectively, and q<sub>t</sub> + q<sub>b</sub> is the total THERMAL POWER in percent RTP.

\*These values denoted with [\*] are specified in the COLR.

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following nominal Trip Setpoint by more than **the following: For ΔT (0.01% ΔT span), for T<sub>avg</sub> (0.01% ΔT span).** ~~[(TBD)]% of ΔT span.~~

$$\Delta T \frac{(1 + \tau_4 s)}{(1 + \tau_5 s)} \leq \Delta T_0 \left\{ K_4 - K_5 \frac{\tau_3 s}{1 + \tau_3 s} T - K_6 [T - T''] - f_2(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.

ΔT<sub>0</sub> is the **loop specific** indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec<sup>-1</sup>.

T is the measured RCS average temperature, °F.

T'' is the **nominal-loop specific indicated** T<sub>avg</sub> at RTP, ≤[\*]°F.

K<sub>4</sub> ≤ [\*]

K<sub>5</sub> ≥ [\*]/°F for increasing T<sub>avg</sub>

K<sub>6</sub> ≥ [\*]/°F

when T > T''

[\*]/°F for decreasing T<sub>avg</sub>

[\*]/°F

when T ≤ T''

τ<sub>3</sub> ≥ [\*] sec

τ<sub>4</sub> ≥ [\*] sec

τ<sub>5</sub> ≤ [\*] sec

f<sub>2</sub>(ΔI) = [\*]

\*These values denoted with [\*] are specified in the COLR.

3.3 INSTRUMENTATION

3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.2 The ESFAS instrumentation for each function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2-1.

ACTIONS

- NOTES -

1. Separate condition entry is allowed for each Function.
2. The Conditions for each Function are given in Table 3.3.2-1. If the Required Actions and associated Completion Times of the first Condition are not met, refer to the second Condition.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels or divisions inoperable.	A.1 Enter the Condition referenced in Table 3.3.2-1 for the channel(s) or division(s).	Immediately
B. One or two channels or divisions inoperable.	B.1 Place one inoperable channel or division in bypass or trip.	[6] hours (7)
	<u>AND</u> B.2 With two inoperable channels or divisions, place one inoperable channel or division in bypass and one inoperable channel or division in trip.	[6] hours (7)
C. One channel inoperable.	C.1 Place inoperable channel in bypass.	[6] hours (7)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One required division inoperable.	D.1 Restore required division to OPERABLE status.	6 hours
E. One switch or switch set inoperable.	E.1 Restore switch and switch set to OPERABLE status.	48 hours
F. One channel inoperable.	F.1 Restore channel to OPERABLE status.	72 hours
	<u>OR</u>	
	F.2.1 Verify alternate radiation monitors are OPERABLE.	72 hours
	<u>AND</u>	
	F.2.2 Verify control room isolation and air supply initiation manual controls are OPERABLE.	72 hours
G. One switch, switch set, channel, or division inoperable.	G.1 Restore switch, switch set, channel, and division to OPERABLE status.	72 hours
H. One channel inoperable.	H.1 Place channel in trip.	6 hours
I. One or two channels inoperable.	I.1 Place one inoperable channel in bypass or trip.	[6] hours (7)
	<u>AND</u>	
	I.2 With two inoperable channels, place one channel in bypass and one channel in trip.	[6] hours (7)
J. One or two interlock channels inoperable.	J.1 Verify the interlocks are in the required state for the existing plant conditions.	1 hour
	<u>OR</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	<p>J.2.1 Place the Functions associated with one inoperable interlock channel in bypass or trip.</p> <p><u>AND</u></p> <p>J.2.2 With two interlock channels inoperable, place the Functions associated with one inoperable interlock channel in bypass and with one inoperable interlock channel in trip.</p>	<p>[7] hours (7)</p> <p>[7] hours (7)</p>
<p>K. Required Action and associated Completion Time not met.</p>	<p>K.1 -----  <b>- NOTE -</b>                      LCO 3.0.8 is not applicable.                      -----</p> <p>Suspend movement of irradiated fuel assemblies.</p>	<p>Immediately</p>
<p>L. Required Action and associated Completion Time not met.</p>	<p>L.1 Be in MODE 3.</p>	<p>6 hours</p>
<p>M. Required Action and associated Completion Time not met.</p>	<p>M.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>M.2 Be in MODE 4.</p>	<p>6 hours</p> <p>12 hours</p>
<p>N. Required Action and associated Completion Time not met.</p>	<p>N.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>N.2 Be in MODE 4 with the RCS cooling provided by the RNS.</p>	<p>6 hours</p> <p>24 hours</p>

SURVEILLANCE REQUIREMENTS

- NOTE -

Refer to Table 3.3.2-1 to determine which SRs apply for each Engineered Safety Features (ESF) Function.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.2.2	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.2.3	<p style="text-align: center;">----- - NOTE - -----</p> <p>Verification of setpoint not required for manual initiation functions.</p> <p style="text-align: center;">-----</p> <p>Perform TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT).</p>	24 months
SR 3.3.2.4	<p style="text-align: center;">----- - NOTE - -----</p> <p>This surveillance shall include verification that the time constants are adjusted to the prescribed values.</p> <p style="text-align: center;">-----</p> <p>Perform CHANNEL CALIBRATION.</p>	24 months
SR 3.3.2.5	Perform CHANNEL OPERATIONAL TEST (COT).	[92] days
SR 3.3.2.6	Verify ESFAS RESPONSE TIMES are within limit.	24 months on a STAGGERED TEST BASIS
SR 3.3.2.7	<p style="text-align: center;">----- - NOTE - -----</p> <p>This Surveillance is not required to be performed for actuated equipment which is included in the Inservice Test (IST) Program.</p> <p style="text-align: center;">-----</p> <p>Perform ACTUATION DEVICE TEST.</p>	24 months
SR 3.3.2.8	Perform ACTUATION DEVICE TEST for squib valves.	24 months

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FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1. Safeguards Actuation						
a. Manual Initiation	1,2,3,4	2 switches	E,O	SR 3.3.2.3	N/ANA	N/ANA
	5	2 switches	G,Y	SR 3.3.2.3	N/ANA	N/ANA
b. Containment Pressure – High 2	1,2,3,4	4	B,O	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 6.21 psig	≤ <del>{8.0}</del> 6.2 psig
c. Pressurizer Pressure – Low	1,2,3 <sup>(a)</sup>	4	B,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 1794.9 psig	≥ <del>{1685}</del> 1795.3 psig
d. Steam Line Pressure – Low	1,2,3 <sup>(a)</sup>	4 per steam line	B,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 559.7 psig <sup>(b)</sup>	≥ <del>{390}</del> 560.3 psig <sup>(b)</sup>
e. RCS Cold Leg Temperature (T <sub>cold</sub> ) – Low	1,2,3 <sup>(a)</sup>	4 per loop	B,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 504.9°F ≤ 505.1°F	≥ <del>{500}</del> 505°F

(a) Above the P-11 (Pressurizer Pressure) interlock, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F.

(b) Time constants used in the lead/lag controller are  $\tau_1 \geq \{50\}$  seconds and  $\tau_2 \leq \{5\}$  seconds.

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Reviewer Note: ~~The values specified in brackets in the Trip Setpoint column are the Chapter 15 safety analysis values and are included for reviewer information only.~~

The values specified in brackets followed by " \* " in the Trip Setpoint and Allowable Value columns are typical values for the Function. No credit was assumed for these Functions (typically diverse trips/actuators) in the Chapter 15 safety analyses and no safety analysis value is available.

~~The "Battery Charger Input Voltage – Low" Functions (15.c and 20.b) value specified is a typical value for the Function. The actual value will depend on the capabilities of the equipment selected with regard to its ability to function with degraded voltage as well as the setpoint methodology.~~

**In all cases, the values specified for trip setpoints and allowable values must be confirmed following completion of the plant-specific setpoint study. Upon selection of the plant specific instrumentation, the Trip Setpoints will be calculated in accordance with the setpoint methodology described in WCAP-16361-P. Allowable Values will be calculated in accordance with the setpoint methodology and specified in the Allowable Value column. The plant specific setpoint calculations will reflect the latest licensing analysis/design basis and may incorporate NRC accepted improvements in setpoint methodology. Following the setpoint study, the values specified in brackets must be replaced with the actual Trip Setpoints. Upon selection of the instrumentation the Trip Setpoints will be calculated in accordance with the setpoint methodology described in WCAP-14606. (WCAP-14606 is an AP600 document that describes a methodology that is applicable to AP1000. AP1000 has some slight differences in instrument spans as a result of the higher power level.) Allowable Values will be calculated in accordance with the setpoint methodology and specified in the Allowable Value column. The setpoint calculations will reflect the design basis and incorporate NRC accepted setpoint methodology.**

~~The Containment Pressure – High 2 setpoint (Functions 1.b, 4.b, and 12.b) should be specified as low as reasonable, without creating potential for spurious trips during normal operations, consistent with the TMI action item (II.E.4.2) guidance.~~

Table 3.3.2-1 (page 2 of 13)  
Engineered Safeguards Actuation System Instrumentation

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FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
2. Core Makeup Tank (CMT) Actuation						
a. Manual Initiation	1,2,3,4 <sup>(j)</sup>	2 switches	E,N	SR 3.3.2.3	N/ANA	N/ANA
	4 <sup>(n)</sup> , 5 <sup>(l)</sup>	2 switches	E,U	SR 3.3.2.3	N/ANA	N/ANA
b. Pressurizer Water Level – Low 2	1,2,3,4 <sup>(j)</sup>	4	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 9.95% span ≤ 10.05% span	≥ [7.0]10% span* ≥ [1.0]10% span
	4 <sup>(n)</sup> , 5 <sup>(l)</sup>	4	B,V	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 9.95% span ≤ 10.05% span	10% span* 10% span ≥ [7.0]10% ± ≥ [1.0]10%
c. Safeguards Actuation	1,2,3,4,5 <sup>(l)</sup>	Refer to Function 1 (Safeguards Actuation) for initiating functions and requirements.				
d. ADS Stages 1, 2, & 3 Actuation	1,2,3,4,5 <sup>(l)</sup>	Refer to Function 9 (ADS Stages 1, 2 & 3 Actuation) for all initiating functions and requirements.				
3. Containment Isolation						
a. Manual Initiation	1,2,3,4	2 switches	E,O	SR 3.3.2.3	N/ANA	N/ANA
	5 <sup>(m)</sup> , 6 <sup>(m)</sup>	2 switches	G,Y	SR 3.3.2.3	N/ANA	N/ANA
b. Manual Initiation of Passive Containment Cooling	1,2,3,4,5 <sup>(e,m)</sup> , 6 <sup>(e,m)</sup>	Refer to Function 12.a (Passive Containment Cooling Actuation) for initiating functions and requirements.				
c. Safeguards Actuation	1,2,3,4,5 <sup>(m)</sup>	Refer to Function 1 (Safeguards Actuation) for initiating functions and requirements.				

(e) With decay heat > 9.0 MWt.

(l) With the RCS pressure boundary intact.

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(m) Not applicable for valve isolation Functions whose associated flow path is isolated.

(n) With the RCS being cooled by the RNS.

Table 3.3.2-1 (page 3 of 13)  
Engineered Safeguards Actuation System Instrumentation

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FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
<b>4. Steam Line Isolation</b>						
a. Manual Initiation	1,2 <sup>(l)</sup> ,3 <sup>(l)</sup> ,4 <sup>(l)</sup>	2 switches	E,S	SR 3.3.2.3	N/ANA	N/ANA
b. Containment Pressure – High 2	1,2 <sup>(l)</sup> ,3 <sup>(l)</sup> ,4 <sup>(l)</sup>	4	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 6.21 psig	≤ <del>{8.0}</del> 6.2 psig
<b>c. Steam Line Pressure</b>						
(1) Steam Line Pressure – Low	1,2 <sup>(l)</sup> ,3 <sup>(a,l)</sup>	4 per steam line	B,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 559.7 psig <sup>(b)</sup>	≥ <del>{390}</del> 560.3 psig <sup>(b)</sup>
(2) Steam Line Pressure – Negative Rate – High	3 <sup>(d,l)</sup>	4 per steam line	B,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 100.1 psig with time constant ≥ 50 seconds	≤ <del>{100}</del> psig with time constant ≥ <del>{50}</del> seconds
d. T <sub>cold</sub> – Low	1,2 <sup>(l)</sup> ,3 <sup>(a,l)</sup>	4 per loop	B,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 504.9°F ≤ 505.1°F	≥ <del>{5050}</del> °F
<b>5. Turbine Trip</b>						
a. Manual Main Feedwater Isolation	1,2	Refer to Function 6.a (Manual Main Feedwater Control Valve Isolation) for requirements.				
b. SG Narrow Range Water Level – High 2	1,2	4 per SG	B,L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 82.05%	≤ <del>{100}</del> 82% span
c. Safeguards Actuation	1,2	Refer to Function 1 (Safeguards Actuation) for initiating functions and requirements.				
d. Reactor Trip	1,2	Refer to Function 18.a (ESFAS Interlocks, Reactor Trip, P-4) for requirements.				

(a) Above the P-11 (Pressurizer Pressure) interlock, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F.

(b) Time constants used in the lead/lag controller are  $\tau_1 \geq \{50\}$  seconds and  $\tau_2 \leq \{5\}$  seconds.

(d) Below the P-11 (Pressurizer Pressure) interlock.

(l) Not applicable if all MSIVs are closed.

Table 3.3.2-1 (page 4 of 13)  
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FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6. Main Feedwater Control Valve Isolation						
a. Manual Initiation	1,2,3,4 <sup>(m)</sup>	2 switches	E,S	SR 3.3.2.3	N/ANA	N/ANA
b. SG Narrow Range Water Level – High 2	1,2,3,4 <sup>(j,m)</sup>	4 per SG	B,R	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 82.05% span	≤ <del>{100}</del> 82% span
c. Safeguards Actuation	1,2,3,4 <sup>(m)</sup>	Refer to Function 1 (Safeguards Actuation) for all initiating functions and requirements.				
d. Reactor Coolant Average Temperature (T <sub>avg</sub> ) – Low 1	1,2	4	B,L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 549.9°F	≥ <del>{542}</del> 550°F
Coincident with Reactor Trip	1,2	Refer to Function 18.a (ESFAS Interlocks, Reactor Trip, P-4) for requirements.				
7. Main Feedwater Pump Trip and Valve Isolation						
a. Manual Initiation	Refer to Function 6.a (Manual Main Feedwater Control Valve Isolation) for requirements.					
b. SG Narrow Range Water Level – High 2	1,2,3,4 <sup>(j,m)</sup>	4 per SG	B,R	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 82.05% span	≤ <del>{100}</del> 82% span
c. Safeguards Actuation	1,2,3,4 <sup>(m)</sup>	Refer to Function 1 (Safeguards Actuation) for all initiating functions and requirements.				
d. Reactor Coolant Average Temperature T <sub>avg</sub> – Low 2	1,2	2 per loop	B,L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 541.9°F*	≥ <del>{542}</del> 542°F*
Coincident with Reactor Trip	1,2	Refer to Function 18.a (ESFAS Interlocks, Reactor Trip, P-4) for requirements.				

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(m) Not applicable for valve isolation Functions whose associated flow path is isolated.

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FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
8. Startup Feedwater Isolation						
a. SG Narrow Range Water Level – High 2	1,2,3,4 <sup>(o)</sup>	4 per SG	B,S	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 82.05% span	≤ <del>[100]</del> 82% span
b. T <sub>cold</sub> – Low	1,2,3 <sup>(a)</sup>	4 per loop	B,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 504.9°F ≤ 505.1°F	≥ <del>[5050]</del> °F±
c. Manual Initiation	Refer to Function 6.a (Manual Main Feedwater Control Valve Isolation) for requirements.					
9. ADS Stages 1, 2 & 3 Actuation						
a. Manual Initiation	1,2,3,4	2 switch sets	E,O	SR 3.3.2.3	N/ANA	N/ANA
	5 <sup>(k)</sup> ,6 <sup>(g,k)</sup>	2 switch sets	G,X	SR 3.3.2.3	N/ANA	N/ANA
b. Core Makeup Tank (CMT) Level – Low 1	1,2,3,4	4 per tank	B,O	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 56.9% span	≥ <del>[67.5]</del> 61.9% volumespan
	5 <sup>(c,k)</sup>	4 per OPERABLE tank	B,V	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 56.9% span	≥ <del>[67.5]</del> 61.9% volumespan
Coincident with CMT Actuation	Refer to Function 2 (CMT Actuation) for all initiating functions and requirements.					

(a) Above the P-11 (Pressurizer Pressure) interlock, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F.

(c) With pressurizer level ≥ 20%.

(g) With upper internals in place.

(o) Not applicable when the startup feedwater flow paths are isolated.

(k) Not applicable when the required ADS valves are open. See LCO 3.4.13 and LCO 3.4.14 for ADS valve and equivalent relief area requirements.

Table 3.3.2-1 (page 6 of 13)  
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
10. ADS Stage 4 Actuation						
a. Manual Initiation Coincident with	1,2,3,4	2 switch sets	E,O	SR 3.3.2.3	N/ANA	N/ANA
	5 <sup>(k)</sup> ,6 <sup>(g,k)</sup>	2 switch sets	G,X	SR 3.3.2.3	N/ANA	N/ANA
RCS Wide Range Pressure – Low, or	1,2,3,4	4	B,O	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 1198 psig	≥{1200} psig
	5 <sup>(k)</sup> ,6 <sup>(g,k)</sup>	4	B,X	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 1198 psig	≥{1200} psig
ADS Stages 1, 2 & 3 Actuation	Refer to Function 9 (Stages 1, 2, & 3 Actuation) for initiating functions and requirements					
b. CMT Level – Low 2	1,2,3,4	4 per tank	B,O	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 56.9% span	61.9% span ≥ {20}% volume level span
	5 <sup>(c,k)</sup>	4 per OPERABLE tank	B,V	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 56.9% span	61.9% span ≥ {20}% volume level span
Coincident with RCS Wide Range Pressure – Low, and	1,2,3,4	4	B,O	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 1198 psig	1200 psig ≥ {1200} psig
	5 <sup>(c,k)</sup>	4	B,V	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 1198 psig	1200 psig ≥ {1200} psig
Coincident with ADS Stages 1, 2 & 3 Actuation	1,2,3,4,5 <sup>(c,k)</sup>	Refer to Function 9 (ADS Stages 1, 2 & 3 Actuation) for initiating functions and requirements				
c. Coincident RCS Loop 1 and 2 Hot Leg Level – Low 2	4 <sup>(n)</sup> ,5 <sup>(k)</sup> ,6 <sup>(k)</sup>	1 per loop	BB,Y	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 2.98 in. above inside surface of the bottom of the hot legs	≥{3} in. above inside surface of the bottom of the hot legs

(c) With pressurizer level ≥ 20%.

(g) With upper internals in place.

(k) Not applicable when the required ADS valves are open. See LCO 3.4.13 and LCO 3.4.14 for ADS valve and equivalent relief area requirements.

Table 3.3.2-1 (page 7 of 13)  
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FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
11. Reactor Coolant Pump Trip						
a. ADS Stages 1, 2 & 3 Actuation	Refer to Function 9 (ADS Stages 1, 2 & 3 Actuation) for initiating functions and requirements.					
b. Reactor Coolant Pump Bearing Water Temperature – High	1,2	4 per RCP	B,L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 230.4°F*	≤ <del>[320]</del> 230°F*
c. Manual CMT Actuation	Refer to Function 2.a (Manual CMT Actuation) for requirements.					
d. Pressurizer Water Level – Low 2	1,2,3,4 <sup>(j)</sup>	4	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 9.95% span ≤ 10.05% span	10% span* 10% span ≥ <del>[7.0]</del> %* ≥ <del>[1.0]</del> %
	4 <sup>(n)</sup> ,5 <sup>(c,i)</sup>	4	B,V	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 9.95% span ≤ 10.05% span	10% span* 10% span ≥ <del>[7.0]</del> %* ≥ <del>[1.0]</del> %
e. Safeguards Actuation	Refer to Function 1 (Safeguards Actuation) for initiating functions and requirements.					
12. Passive Containment Cooling Actuation						
a. Manual Initiation	1,2,3,4	2 switches	E,O	SR 3.3.2.3	N/ANA	N/ANA
	5 <sup>(e)</sup> ,6 <sup>(e)</sup>	2 switches	G,Y	SR 3.3.2.3	N/ANA	N/ANA
b. Containment Pressure – High 2	1,2,3,4	4	B,O	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 6.21 psig	≤ <del>[8.0]</del> 6.2 psig

(c) With pressurizer level ≥ 20%.

(e) With decay heat > 9.0 MWt.

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(n) With the RCS being cooled by the RNS.

Table 3.3.2-1 (page 8 of 13)  
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FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
13. Passive Residual Heat Removal Heat Exchanger Actuation						
a. Manual Initiation	1,2,3,4	2 switches	E,O	SR 3.3.2.3	N/ANA	N/ANA
	5 <sup>(l)</sup>	2 switches	E,U	SR 3.3.2.3	N/ANA	N/ANA
b. SG Narrow Range Water Level – Low	1,2,3,4 <sup>(j)</sup>	4 per SG	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 20.95% span	≥ {95,000} lbm 21% span
Coincident with Startup Feedwater Flow – Low	1,2,3,4 <sup>(j)</sup>	2 per feedwater line	H,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 198.8 gpm per SG	≥ {200} gpm per SG <sup>±</sup>
c. SG Wide Range Water Level – Low	1,2,3,4 <sup>(j)</sup>	4 per SG	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 53.95% span	≥ {55,000} lbm 54% span
d. ADS Stages 1, 2 & 3 Actuation	1,2,3,4,5 <sup>(l)</sup>	Refer to Function 9 (ADS Stages 1, 2 & 3 Actuation) for initiating functions and requirements.				
e. CMT Actuation	Refer to Function 2 (CMT Actuation) for initiating functions and requirements.					
f. Pressurizer Water Level, High 3	1,2,3,4 <sup>(j,p)</sup>	4	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 71.05%	≤ {80} 71% <sup>±</sup>

(l) With the RCS pressure boundary intact.

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(p) Above the P-19 (RCS Pressure) interlock.



Table 3.3.2-1 (page 9 of 13)  
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FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
14. SG Blowdown Isolation						
a. Passive Residual Heat Removal Heat Exchanger Actuation	1,2,3,4 <sup>(j,m)</sup>	Refer to Function 13 (Passive Residual Heat Removal Heat Exchanger Actuation) for all initiating functions and requirements.				
b. SG Narrow Range Water Level – Low	1,2,3,4 <sup>(j)</sup>	4 per SG	B,R	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 20.95% span	≥ {95,000} <del>16m</del> 21% span
15. Boron Dilution Block						
a. Source Range Neutron Flux Multiplication	2 <sup>(f)</sup> ,3,4 <sup>(m)</sup>	4	B,T	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ Source Range Flux X 1.601 in 50 minutes	≤ Source Range Flux X {1.6 in 50} minutes
	5 <sup>(m)</sup>	4	B,P	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ Source Range Flux X 1.601 in 50 minutes	≤ Source Range Flux X {1.6 in 50} minutes
b. Reactor Trip	Refer to Function 18.a (ESFAS Interlocks, Reactor Trip, P-4) for all requirements.					
c. Battery Charger Input Voltage – Low	1,2,3,4 <sup>(m)</sup>	4 divisions	B,T	SR 3.3.2.3 SR 3.3.2.4	≥ 342.6 V	≥ {3453} V±
	5 <sup>(m)</sup>	4 divisions	B,P	SR 3.3.2.3 SR 3.3.2.4	≥ 342.6 V	≥ {3453} V±
16. Chemical Volume and Control System Makeup Isolation						
a. SG Narrow Range Water Level – High 2	1,2,3 <sup>(m)</sup> ,4 <sup>(j,m)</sup>	4 per SG	B,R	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 82.05% span	≤ {100} 82% span
b. Pressurizer Water Level – High 1	1,2,3 <sup>(m)</sup>	4	B,Q	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 23.05% span	≤ {30} 23%± span
	1,2,3 <sup>(m)</sup>	Refer to Function 1 (Safeguards Actuation) for initiating functions and requirements.				
c. Pressurizer Water Level – High 2	1,2,3,4 <sup>(j,m,p)</sup>	4	B,T	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 59.05% span	≤ {67} 59% span
d. Containment Radioactivity – High 2	1,2,3 <sup>(m)</sup>	4	B,Q	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 150 R/hr	≤ {100} R/hr
e. Manual Initiation	1,2,3 <sup>(m)</sup> ,4 <sup>(j,m)</sup>	2 switches	E,R	SR 3.3.2.3	N/ANA	N/ANA

(f) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(m) Not applicable for valve isolation Functions whose associated flow path is isolated.

(p) Above the P-19 (RCS Pressure) interlock.

Table 3.3.2-1 (page 10 of 13)  
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FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
17. Normal Residual Heat Removal System Isolation						
a. Containment Radioactivity – High 2	1,2,3 <sup>(m)</sup>	4	B,Q	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 150 R/hr	100 R/hr ≤ {100} R/hr
b. Safeguards Actuation	1,2,3 <sup>(m)</sup>	Refer to Function 1 (Safeguards Actuation) for all initiating functions and requirements.				
c. Manual Initiation	1,2,3 <sup>(m)</sup>	2 switch sets	E,Q	SR 3.3.2.3	N/ANA	N/ANA
18. ESFAS Interlocks						
a. Reactor Trip, P-4	1,2,3	3 divisions	D,M	SR 3.3.2.3	N/ANA	N/ANA
b. Pressurizer Pressure, P-11	1,2,3	4	J,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 1970.4 psig	≤ {1970} psig
c. Intermediate Range Neutron Flux, P-6	2	4	J,L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 9.91 E-6% RTP	≥ {1E-405}% amps RTP
d. Pressurizer Level, P-12	1,2,3,4,5,6	4	J,M BB,Y	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 16.05% span	Above Pressurizer Water Level – Low 4 setpoint of {20} 16% span
e. RCS Pressure, P-19	1,2,3,4 <sup>(j)</sup>	4	J,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 702 psig	≥ {700} psig
19. Containment Air Filtration System Isolation						
a. Containment Radioactivity – High 1	1,2,3,4 <sup>(j)</sup>	4	B,Z	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 3 R/hr	≤ {2} R/hr
b. Containment Isolation	Refer to Function 3 (Containment Isolation) for initiating functions and requirements.					

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(m) Not applicable for valve isolation Functions whose associated flow path is isolated.

Table 3.3.2-1 (page 11 of 13)  
Engineered Safeguards Actuation System Instrumentation

8

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
20. Main Control Room Isolation and Air Supply Initiation						
a. Control Room Air Supply Radiation – High 2	1,2,3,4	2	F,O	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	$\leq 1.5 \times 10^{-6}$ curies/m <sup>3</sup> <b>DOSE EQUIVALENT I-131</b>	$\leq [21 \times 10^{-6}]$ curies/m <sup>3</sup> <b>DOSE EQUIVALENT I-131</b>
	Note (h)	2	G,K	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	$\leq 1.5 \times 10^{-6}$ curies/m <sup>3</sup> <b>DOSE EQUIVALENT I-131</b>	$\leq [21 \times 10^{-6}]$ curies/m <sup>3</sup> <b>DOSE EQUIVALENT I-131</b>
b. Battery Charger Input Voltage – Low	1,2,3,4	4 divisions	B,O	SR 3.3.2.3 SR 3.3.2.4	$\geq 342.6$ V	$\geq [343]345$ V <sup>±</sup>
	Note (h)	4 divisions	G,K	SR 3.3.2.3 SR 3.3.2.4	$\geq 342.6$ V	$\geq [343]345$ V <sup>±</sup>
21 Auxiliary Spray and Purification Line Isolation						
a. Pressurizer Water Level – Low 1	1,2	4	B,L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	$\geq 19.95\%$ span	$[20.0]\%$ ± span
b. Manual Initiation	1,2	Refer to Function 16.e (Manual Chemical Volume Control System (Makeup Isolation)) for requirements.				
22. In-Containment Refueling Water Storage Tank (IRWST) Injection Line Valve Actuation						
a. Manual Initiation	1,2,3,4 <sup>(j)</sup>	2 switch sets	E,N	SR 3.3.2.3	N/ANA	N/ANA
	4 <sup>(n)</sup> ,5,6	2 switch sets	G,Y	SR 3.3.2.3	N/ANA	N/ANA
b. ADS 4th Stage Actuation	Refer to Function 10 (ADS 4th Stage Actuation) for initiating functions and requirements.					
c. Coincident RCS Loop 1 and 2 Hot Leg Level – Low 2	4 <sup>(n)</sup> ,5,6	1 per loop	BB,Y	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	$\geq 2.98$ in. above inside surface of the bottom of the hot legs	$\geq [3]$ in. above inside surface of the bottom of the hot legs

(h) During movement of irradiated fuel assemblies.

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(n) With the RCS being cooled by the RNS.

Table 3.3.2-1 (page 12 of 13)  
Engineered Safeguards Actuation System Instrumentation

8

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
23. IRWST Containment Recirculation Valve Actuation						
a. Manual Initiation	1,2,3,4 <sup>(j)</sup>	2 switch sets	E,N	SR 3.3.2.3	N/ANA	N/ANA
	4 <sup>(n)</sup> ,5,6	2 switch sets	G,Y	SR 3.3.2.3	N/ANA	N/ANA
b. ADS Stage 4 Actuation	Refer to Function 10 (ADS Stage 4 Actuation) for all initiating functions and requirements.					
Coincident with IRWST Level – Low 3	1,2,3,4 <sup>(j)</sup>	4	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ Containment Elevation @ 109.99 ft.	≥ Containment Elevation @ [11007 ft.'2"]
	4 <sup>(n)</sup> ,5 <sup>(k)</sup> ,6 <sup>(k)</sup>	4	I,Y	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ Containment Elevation @ 109.99 ft.	Containment Elevation @ 110 ft. ≥ Containment Elevation @ [107'2"]
24. Refueling Cavity Isolation						
a. Spent Fuel Pool Level – Low	6	3	H,P	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 37.49 ft.	[37.5] ft.
25. ESF Coincidence Logic						
a. Coincidence Logic	1,2,3,4	4 divisions, 1 battery-backed subsystem per division	D,O	SR 3.3.2.2	N/ANA	N/ANA
	5,6	4 divisions, 1 battery-backed subsystem per division	G,W	SR 3.3.2.2	N/ANA	N/ANA

(k) Not applicable when the required ADS valves are open. See LCO 3.4.13 and LCO 3.4.14 for ADS valve and equivalent relief area requirements.

Table 3.3.2-1 (page 13 of 13)  
Engineered Safeguards Actuation System Instrumentation

8

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
26. ESF Actuation						
a. ESF Actuation Subsystem	1,2,3,4	4 divisions, 1 battery-backed subsystem per division	D,O	SR 3.3.2.2 SR 3.3.2.7 SR 3.3.2.8	N/ANA	N/ANA
	5,6	4 divisions, 1 battery-backed subsystem per division	G,W	SR 3.3.2.2 SR 3.3.2.7	N/ANA	N/ANA
27. Pressurizer Heater Trip						
a. Core Makeup Tank Actuation	1,2,3,4 <sup>(j,p)</sup>	Refer to Function 2 (Core Makeup Tank Actuation) for all initiating functions and requirements. In addition to the requirements for Function 2, SR 3.3.2.9 also applies.				
b. Pressurizer Water Level, High 3	1,2,3,4 <sup>(j,p)</sup>	4	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≤ 71.05%	<del>{80}</del> 71%*
28. Chemical and Volume Control System Letdown Isolation						
a. Hot Leg Level – Low 1	4 <sup>(n)</sup> ,5,6 <sup>(q)</sup>	1 per loop	C,AA	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 17.98 in. above inside surface of the bottom of the hot legs	≥ <del>{18}</del> in. above inside surface of the bottom of the hot legs
29. SG Power Operated Relief Valve and Block Valve Isolation						
a. Manual Initiation	1,2,3,4 <sup>(j)</sup>	2 switches	E,N	SR 3.3.2.3	N/ANA	N/ANA
b. Steam Line Pressure – Low	1,2,3,4 <sup>(j)</sup>	4 per steam line	B,N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6	≥ 559.7 psig	<del>{390}</del> 560.3 psig <sup>(b)</sup>

1

(b) Time constants used in the lead/lag controller are  $\tau_1 \geq \{50\}$  seconds and  $\tau_2 \leq \{5\}$  seconds.

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(m) Not applicable for valve isolation Functions whose associated flow path is isolated.

(n) With the RCS being cooled by the RNS.

(p) Above the P-19 (RCS Pressure) interlock.

(q) With the water level < 23 feet above the top of the reactor vessel flange.

3.4 REACTOR COOLANT SYSTEM (RCS)

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

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

- a. Pressurizer Pressure is greater than or equal to the limit specified in the COLR
- b. RCS Average Temperature is less than or equal to the limit specified in the COLR, and
- c. RCS total flow rate  $\geq$  {301,670} gpm and greater than or equal to the limit specified in the COLR.

1

APPLICABILITY: MODE 1.

- NOTE -

Pressurizer pressure limit does not apply during:

- a. THERMAL POWER ramp > 5% RTP per minute, or
- b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION		REQUIRED ACTION	COMPLETION TIME
A.	One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B.	Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is less than or equal to the limit specified in the COLR.	12 hours
1   SR 3.4.1.3	Verify RCS total flow rate is $\geq$ {301,670} gpm and greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.4	----- - NOTE - Not required to be performed until 24 hours after $\geq$ 90% RTP. -----	24 months
1	Verify by precision heat balance that RCS total flow rate is $\geq$ {301,670} gpm and greater than or equal to the limit specified in the COLR.	

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

6

LCO 3.4.2 Each RCS loop average temperature ( $T_{avg}$ ) shall be  $\geq \{551\}^{\circ}\text{F}$ .

APPLICABILITY: MODE 1,  
MODE 2 with  $k_{eff} \geq 1.0$ .

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. $T_{avg}$ in one or more RCS loops not within limit.	A.1 Be in MODE 2 with $k_{eff} < 1.0$ .	30 minutes

SURVEILLANCE REQUIREMENTS

6

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS $T_{avg}$ in each loop $\geq \{551\}^{\circ}\text{F}$ .	12 hours



3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops

LCO 3.4.4 Two RCS loops shall be OPERABLE and in operation (Four Reactor Coolant Pumps (RCPs) operating with variable speed control bypassed).

- NOTES -

1

1. No RCP shall be started when the reactor trip breakers are closed.
2. **No RCP shall be started when the RCS temperature is  $\geq 200^{\circ}\text{F}$  unless pressurizer level is  $< 92\%$ .**
32. No RCP shall be started with any RCS cold leg temperature  $\leq [20075]^{\circ}\text{F}$  unless the secondary side water temperature of each steam generator (SG) is  $\leq [50]^{\circ}\text{F}$  above each of the RCS cold leg temperatures.
34. All RCPs may be de-energized in MODE 3, 4, or 5 for  $\leq 1$  hour per 8 hour period provided:
  - a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and
  - b. Core outlet temperature is maintained at least  $10^{\circ}\text{F}$  below saturation temperature.

APPLICABILITY: MODES 1 and 2, MODES 3, 4, and 5, whenever the reactor trip breakers are closed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----- <b>- NOTE -</b> Required Action A.1 must be completed whenever Condition A is entered. ----- Requirements of LCO not met in MODE 1 or 2.	A.1 Be in MODE 3 with the reactor trip breakers open.	6 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 Minimum RCS Flow

LCO 3.4.8

12

At least one Reactor Coolant Pump (RCP) shall be in operation with a total flow through the core of at least [403,000] gpm.

- NOTES -

1. All RCPs may be de-energized for  $\leq 1$  hour per 8 hour period provided:
  - a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and
  - b. Core outlet temperature is maintained at least 10°F below saturation temperature.
2. **No RCP shall be started when the RCS temperature is  $\geq 200^\circ\text{F}$  unless pressurizer level is  $< 92\%$ .**
23. No RCP shall be started with any RCS cold leg temperature  $\leq [20075]^\circ\text{F}$  unless the secondary side water temperature of each steam generator (SG) is  $\leq [50]^\circ\text{F}$  above each of the RCS cold leg temperatures.

1

APPLICABILITY: MODES 3, 4, and 5, whenever the reactor trip breakers are open.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. No RCP in operation.	A.1 Isolate all sources of unborated water.	1 hour
	<u>AND</u>	
	A.2 Perform SR 3.1.1.1, (SDM verification).	1 hour

SURVEILLANCE REQUIREMENTS

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SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify that at least one RCP is in operation at $\geq$ [1025]% rated speed or equivalent.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.14 At least one of the following Overpressure Protection Systems shall be OPERABLE, with the accumulators isolated:

13

- a. The Normal Residual Heat Removal System (RNS) suction relief valve, or
- b. The RCS depressurized and an RCS vent of  $\geq [9.3]$  square inches.

- NOTE -

When the RCS temperature is  $\geq 200^\circ\text{F}$ , a reactor coolant pump (RCP) may not be started if the pressurizer level is  $\geq 92\%$ .

APPLICABILITY: MODE 4 when any cold leg temperature is  $\leq 275^\circ\text{F}$ ,  
MODE 5,  
MODE 6 when the reactor vessel head is on.

- NOTE -

Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. An accumulator not isolated when the accumulator pressure is $\geq$ to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	A.1 Isolate affected accumulator.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Increase RCS cold leg temperature to a level acceptable for the existing accumulator pressure allowed in the PTLR.	12 hours
	<u>OR</u>	
	B.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours
C. The RNS suction relief valve inoperable.	C.1 Restore the RNS suction relief valve to OPERABLE status.	12 hours
	<u>OR</u>	
	C.2 Depressurize RCS and establish RCS vent of $\geq [9.3]$ square inches.	12 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	Verify each accumulator is isolated.	12 hours
SR 3.4.14.2	Verify both RNS suction isolation valves in one RNS suction flow path are open.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.14.3	<p style="text-align: center;">-----  <b>- NOTE -</b>                      Only required to be performed when complying with                      LCO 3.4.14.b.                      -----</p> <p>Verify RCS vent <math>\geq</math> [9.3] square inches is open.</p>	<p>12 hours for unlocked-open vent</p> <p><u>AND</u></p> <p>31 days for locked-open vent</p>
SR 3.4.14.4	Verify the lift setting of the RNS suction relief valve.	In accordance with the Inservice Testing Program

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2 Verify a second OPERABLE PIV can meet the leakage limits. This valve is required to be a check valve, or a closed valve, if it isolates a line that penetrates containment.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours  36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.15.1 <b>18</b> Verify leakage of each RCS PIV is equivalent to $\leq 0.5$ gpm per nominal inch valve size up to a maximum of 5 gpm at an RCS pressure $\geq$ {2215} and $\leq$ {2255} psig.	24 months

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time not met.  <u>OR</u>  LCO not met for reasons other than A, B, C, D, or E.	F.1 Be in MODE 3.	6 hours
	<u>AND</u>  F.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.5.2.1 Verify the temperature of the borated water in each CMT is < 120°F.	24 hours
SR 3.5.2.2 Verify the borated water volume in each CMT is ≥ 2500 cu. ft.	7 days
SR 3.5.2.3 Verify each CMT inlet isolation valve is fully open.	12 hours
1 SR 3.5.2.4 Verify the volume of noncondensable gases in each CMT inlet line is ≤ {0.2} ft <sup>3</sup> .	24 hours
SR 3.5.2.5 Verify the boron concentration in each CMT is ≥ 3400 ppm, and ≤ 3700 ppm.	7 days
SR 3.5.2.6 Verify each CMT outlet isolation valve is OPERABLE by stroking it open.	In accordance with the Inservice Testing Program
SR 3.5.2.7 Verify system flow performance of each CMT in accordance with the System Level OPERABILITY Testing Program.	10 years



SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify the outlet manual isolation valve is fully open.	12 hours
SR 3.5.4.2	Verify the inlet motor operated isolation valve is open.	12 hours
SR 3.5.4.3	Verify the volume of noncondensable gases in the PRHR HX inlet line is $\leq \{0.94\} \text{ ft}^3$ .	24 hours
SR 3.5.4.4	Verify that power is removed from the inlet motor operated isolation valve.	31 days
SR 3.5.4.5	Verify both PRHR air operated outlet isolation valves and both IRWST gutter isolation valves are OPERABLE by stroking open the valves.	In accordance with the System Level Inservice Testing Program
SR 3.5.4.6	Verify PRHR HX heat transfer performance in accordance with the System Level OPERABILITY Testing Program.	10 years
SR 3.5.4.7	Verify by visual inspection that the IRWST gutters are not restricted by debris.	24 months

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. One motor operated IRWST isolation valve not fully open.</p> <p><u>OR</u></p> <p>Power is not removed from one or more motor operated IRWST isolation valves.</p>	<p>C.1 Restore motor operated IRWST isolation valve to fully open condition with power removed from both valves.</p>	<p>1 hour</p>
<p>D. Required Action and associated Completion Time not met.</p> <p><u>OR</u></p> <p>LCO not met for reasons other than A, B, or C.</p>	<p>D.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>D.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.5.6.1 Verify the IRWST water temperature is &lt; 120°F.</p>	<p>24 hours</p>
<p>SR 3.5.6.2 Verify the IRWST borated water volume is &gt; {73,9100} cu. ft.</p>	<p>24 hours</p>
<p>SR 3.5.6.3 Verify the IRWST boron concentration is ≥ 2600 ppm and ≤ 2900 ppm.</p>	<p>31 days</p> <p><u>AND</u></p> <p>Once within 6 hours after each solution volume increase of 15,000 gal</p>

1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.  <u>OR</u>  LCO not met for reasons other than A, B, or C.	D.1 Initiate action to be in MODE 6 with the water level $\geq$ 23 feet above the top of the reactor vessel flange.  <u>AND</u>  D.2 Suspend positive reactivity additions.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.5.8.1 Verify the IRWST and refueling cavity water temperature is $<$ 120°F.	24 hours
SR 3.5.8.2 Verify the IRWST and refueling cavity water total borated water volume is $>$ [73,9100] cu. ft.	24 hours
SR 3.5.8.3 Verify the IRWST and refueling cavity boron concentration is $\geq$ 2600 ppm and $\leq$ 2900 ppm.	31 days  <u>AND</u>  Once within 6 hours after each solution volume increase of 15,000 gal
SR 3.5.8.4 For the IRWST and flow paths required to be OPERABLE, the following SRs of Specification 3.5.6, "In-containment Refueling Water Storage Tank (IRWST) – Operating" are applicable:  SR 3.5.6.4 SR 3.5.6.6 SR 3.5.6.8 SR 3.5.6.5 SR 3.5.6.7	In accordance with applicable SRs

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3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

1

LCO 3.6.4 Containment pressure shall be  $\geq$  [-0.2] psig and  $\leq$  +1.0 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4.1 Verify containment pressure is within limits.	12 hours

**- Reviewer's Note -**

The low pressure limit is not needed for plant locations for which the lowest possible ambient temperature is approximately 20°F.

3.6 CONTAINMENT SYSTEMS

3.6.8 Containment Penetrations

LCO 3.6.8 The containment penetrations shall be in the following status:

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- a. The equipment hatches closed and held in place by {four} bolts or, if open, clear of obstructions such that the hatches can be closed prior to steaming into the containment.
- b. One door in each air lock closed or, if open, the containment air locks shall be clear of obstructions such that they can be closed prior to steaming into the containment.
- c. The containment spare penetrations, if open, shall be clear of obstructions such that the penetrations can be closed prior to steaming into the containment.
- d. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:
  - 1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or
  - 2. capable of being closed by an OPERABLE Containment Isolation signal.

APPLICABILITY: MODES 5 and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Restore containment penetrations to required status.	1 hour

3.6 CONTAINMENT SYSTEMS

3.6.9 pH Adjustment

- 7 | LCO 3.6.9 The pH adjustment baskets shall contain  $\geq$  {560} ft<sup>3</sup> of trisodium phosphate (TSP).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The volume of trisodium phosphate not within limit.	A.1 Restore volume of trisodium phosphate to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
7   SR 3.6.9.1 Verify that the pH adjustment baskets contain at least {560} ft <sup>3</sup> of TSP (Na <sub>3</sub> PO <sub>4</sub> -12 H <sub>2</sub> O).	24 months
SR 3.6.9.2 Verify that a sample from the pH adjustment baskets provides adequate pH adjustment of the post-accident water.	24 months

Table 3.7.1-1 (page 1 of 1)  
OPERABLE MSSVs versus Maximum Allowable Power

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (% RTP)
5	[82]
4	[65]
3	[48]
2	[31]

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Conditions A, B, or C not met during movement of irradiated fuel.	E.1 Suspend movement of irradiated fuel assemblies.	Immediately
F. VES inoperable in MODE 1, 2, 3, or 4.	F.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	F.2 Be in MODE 4.	12 hours
	<u>AND</u>	
	F.3 Restore VES to OPERABLE status.	36 hours
G. VES inoperable during movement of irradiated fuel.	G.1 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
	SR 3.7.6.1 Verify Main Control Room air temperature is $\leq 75^{\circ}\text{F}$ .	24 hours
1	SR 3.7.6.2 Verify that the compressed air storage tanks are pressurized to $\geq [3400]$ psig.	24 hours
	SR 3.7.6.3 Verify that each VES air delivery isolation valve is OPERABLE.	In accordance with the Inservice Testing Program
	SR 3.7.6.4 Verify that each VES air header manual isolation valve is in an open position.	31 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 DC Sources – Operating

LCO 3.8.1 The Division A, B, C, and D Class 1E DC power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>11</p> <p>A. One or more battery chargers in one division inoperable.</p>	<p>A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage.</p> <p><u>AND</u></p>	6 hours
	<p>A.2 Verify battery float current ≤ [5] amps.</p> <p><u>AND</u></p>	Once per 24 hours
	<p>A.3 Restore battery charger(s) to OPERABLE status.</p>	7 days
<p>11</p> <p>B. One or more battery chargers in two divisions inoperable.</p>	<p>B.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage.</p> <p><u>AND</u></p>	2 hours
	<p>B.2 Verify battery float current ≤ [5] amps.</p> <p><u>AND</u></p>	Once per 24 hours
	<p>B.3 Restore battery charger(s) to OPERABLE status.</p>	

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.2	<p>Verify each battery charger supplies <math>\geq</math> {400} amps at greater than or equal to the minimum established float voltage for <math>\geq</math> {8} hours.</p> <p style="text-align: center;">①</p> <p style="text-align: center;"><u>OR</u></p> <p>Verify each battery charger can recharge the battery to the fully charged state within {24} hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.</p>	24 months
SR 3.8.1.3	<p style="text-align: center;">----- - NOTES - -----</p> <ol style="list-style-type: none"> <li>1. The modified performance discharge test in SR 3.8.7.6 may be performed in lieu of SR 3.8.1.3.</li> <li>2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4 unless the spare battery is connected to replace the battery being tested. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced.</li> </ol> <p style="text-align: center;">-----</p> <p>Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.</p>	24 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Battery Parameters

LCO 3.8.7 Battery Parameters for Division A, B, C, and D batteries shall be within limits.

APPLICABILITY: When associated DC electrical power sources are required to be OPERABLE.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<span style="border: 1px solid black; border-radius: 50%; padding: 2px;">1</span>   A. One or more batteries in one division with one or more battery cells float voltage < [2.07] V.	A.1 Perform SR 3.8.1.1.	2 hours
	<u>AND</u>	
	A.2 Perform SR 3.8.7.1.	2 hours
<span style="border: 1px solid black; border-radius: 50%; padding: 2px;">1</span>   A.3 Restore affected cell voltage ≥ [2.07] V.	<u>AND</u>	
	A.3 Restore affected cell voltage ≥ [2.07] V.	24 hours
<span style="border: 1px solid black; border-radius: 50%; padding: 2px;">11</span>   B. One or more batteries in one division with float current > [5] amps.	B.1 Perform SR 3.8.1.1.	2 hours
	<u>AND</u>	
	B.2 Restore battery float current to ≤ [5] amps.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>F. Required Action and associated Completion Time not met.</p> <p><u>OR</u></p> <p>One or more batteries in one division with one or more battery cells float voltage &lt; {2.07} V and float current &gt; [5] amps.</p>	<p>F.1 Declare associated battery inoperable.</p>	<p>Immediately</p>

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.8.7.1 -----</p> <p style="text-align: center;"><b>- NOTE -</b></p> <p style="text-align: center;">Not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.1.1.</p> <p>-----</p> <p>Verify each battery float current is ≤ [5] amps.</p>	<p>7 days</p>
<p>SR 3.8.7.2 Verify each battery pilot cell voltage is ≥ {2.07} V.</p>	<p>31 days</p>
<p>SR 3.8.7.3 Verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.</p>	<p>31 days</p>
<p>SR 3.8.7.4 Verify each battery pilot cell temperature is greater than or equal to minimum established design limits.</p>	<p>31 days</p>
<p>SR 3.8.7.5 Verify each battery connected cell voltage is ≥ {2.07} V.</p>	<p>92 days</p>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.7.6</p> <p style="text-align: center;">-----  <b>- NOTE -</b>            This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR.            -----</p> <p>Verify battery capacity is <math>\geq</math> {80}% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.</p> <p style="text-align: center;">①</p>	<p>60 months</p> <p><u>AND</u></p> <p>12 months when battery shows degradation, or has reached {85}% of the expected life with capacity &lt; 100% of manufacturer's rating</p> <p><u>AND</u></p> <p>24 months when battery has reached {85}% of the expected life with capacity <math>\geq</math> 100% of manufacturer's rating</p>

3.9 REFUELING OPERATIONS

3.9.5 Containment Penetrations

LCO 3.9.5 The containment penetrations shall be in the following status:

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- a. The equipment hatches closed and held in place by {four} bolts or, if open, the containment air filtration system (VFS) shall be OPERABLE and operating;
- b. One door in each air lock closed or, if open, the VFS shall be OPERABLE and operating;
- c. The containment spare penetrations closed or, if open, the VFS shall be OPERABLE and operating;
- d. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:
  - 1. Closed by a manual or automatic isolation valve, blind flange, or equivalent, or
  - 2. Capable of being closed by an OPERABLE Containment Isolation signal.

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- NOTE -  
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Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.  
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APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

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- NOTE -  
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LCO 3.0.8 is not applicable.  
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CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

**SURVEILLANCE REQUIREMENTS**

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.5.2	<p style="text-align: center;">-----  <b>- NOTE -</b>                      Not required to be met for containment purge and exhaust valve(s) in penetrations closed to comply with LCO 3.9.4.d.1.                      -----</p> <p>Verify each required containment purge and exhaust valve actuates to the isolation position on a manual actuation signal.</p>	In accordance with the Inservice Test Program
SR 3.9.5.3	Verify the VFS can maintain a negative pressure ( $\leq$ [-0.125] inches water gauge relative to outside atmospheric pressure) in the area enclosed by the containment and alternate barrier.	24 months
SR 3.9.5.4	Operate each VFS train for $\geq$ 10 continuous hours with the heaters operating.	Within 31 days prior to fuel movement or CORE ALTERATIONS

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3.9 REFUELING OPERATIONS

3.9.6 Containment Air Filtration System (VFS)

LCO 3.9.6 One VFS exhaust subsystem shall be OPERABLE.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel building.

ACTIONS

- NOTE -

LCOs 3.0.3 and 3.0.8 are not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required VFS exhaust subsystem inoperable.	A.1 Suspend movement of irradiated fuel assemblies in the fuel building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.6.1 Operate each VFS exhaust subsystem for $\geq 10$ continuous hours with the heaters operating.	Within 31 days prior to fuel movement
SR 3.9.6.2 Verify the VAS fuel handling area subsystem aligns to the VFS exhaust subsystem on an actual or simulated actuation signal.	24 months
SR 3.9.6.3 Verify one VFS exhaust subsystem can maintain a negative pressure ( $\leq \{-0.125\}$ inches water gauge relative to outside atmospheric pressure) in the fuel handling area.	24 months

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## 4.0 DESIGN FEATURES

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### 4.1 Site

[Not applicable to AP1000 Design Certification. Site specific information to be provided by COL Applicant.]

#### 4.1.1 Site and Exclusion Boundaries

[This information will be provided by the combined license applicant.]

#### 4.1.2 Low Population Zone (LPZ)

[This information will be provided by the combined license applicant.]

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### 4.2 Reactor Core

#### 4.2.1 Fuel Assemblies

The reactor shall contain 157 fuel assemblies. Each assembly shall consist of a matrix of fuel rods clad with a zirconium based alloy and containing an initial composition of natural or slightly enriched uranium dioxide (UO<sub>2</sub>) as fuel material. Limited substitutions of zirconium based 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 staff approved codes and methods and shown by tests or analyses to comply with 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 and Gray Rod Assemblies

The reactor core shall contain 53 Rod Cluster Control Assemblies (RCCAs), each with 24 rodlets/RCCA. The RCCA absorber material shall be silver indium cadmium as approved by the NRC.

Additionally, there are 16 low worth Gray Rod Cluster Assemblies (GRCA), with 24 rodlets/GRCA, which, in conjunction with the RCCAs, are used to augment MSHIM load follow operation.

## 4.0 DESIGN FEATURES

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### 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.  $k_{\text{eff}} \leq 0.95$  if fully flooded with unborated water which includes an allowance for uncertainties as described in Section 9.1, "Fuel Storage and Handling."
- c. A nominal {10.90} inch center-to-center distance between fuel assemblies placed in the spent fuel storage racks.

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.  $k_{\text{eff}} \leq 0.95$  if fully flooded with unborated water which includes an allowance for uncertainties as described in Section 9.1, "Fuel Storage and Handling."
- c.  $k_{\text{eff}} \leq 0.98$  if moderated by aqueous foam which includes an allowance for uncertainties as described in Section 9.1, "Fuel Storage and Handling."
- d. A nominal {10.90} inch center-to-center distance between fuel assemblies placed in the new fuel storage racks.

#### 4.3.2 Drainage

The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below a minimum water depth of  $\geq 23$  ft above the surface of the fuel storage racks.

#### 4.3.3 Capacity

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The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than [616] fuel assemblies.

## 5.0 ADMINISTRATIVE CONTROLS

### 5.1 Responsibility

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- 5.1.1 The [Plant Manager] shall be responsible for overall unit operations and shall delegate in writing the succession to this responsibility during his absence.
- 17 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 (SS)] shall be responsible for the control room command function. During any absence of the [SS] from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function.
- 17 During any absence of the [SS] from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.
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## 5.0 ADMINISTRATIVE CONTROLS

### 5.2 Organization

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#### 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.

- 17
- 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; and
  - 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 operation pressures.

#### 5.2.2 Unit Staff

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**- REVIEWER'S NOTE -**

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[Determination of the unit staff positions, numbers, and qualifications are the responsibility of the COL applicant. Input provided in WCAP-14694, Revision 0, for the MCR staff and WCAP-14655, Revision 1, for other than the MCR staff will be used in the determination. Each of the following paragraphs may need to be corrected to specify the plant staffing requirements.]  
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5.2 Organization

5.2.2 Unit Staff (continued)

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The unit staff organization shall include the following:

- 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 5.2.2.a and 5.2.2.g 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.
- 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 procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed Senior Reactor Operators (SROs), licensed Reactor Operators (ROs), 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 the plant manager's 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 be assigned.

- e. The operations manager or assistant operations manager shall hold an SRO license.
- f. 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

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- REVIEWER'S NOTE -

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[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.]

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Each member of the unit staff shall meet or exceed the minimum qualifications of [Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standards acceptable to the NRC staff]. The staff not covered by [Regulatory Guide 1.8] shall meet or exceed the minimum qualifications of [Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff].

5.3.2

For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

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## 5.0 ADMINISTRATIVE CONTROLS

### 5.4 Procedures

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5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

- a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;
  - b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in {Generic Letter 82-33};
  - c. Quality assurance for effluent and environmental monitoring;
  - d. Fire Protection Program implementation; and
  - e. All programs specified in Specification 5.5.
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## 5.5 Programs and Manuals

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### 5.5.7 Safety Function Determination Program (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

### 5.5.8 Containment Leakage Rate Testing Program

- a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995," as modified by approved exceptions.
- b. The calculated peak containment internal pressure for the design basis loss of coolant accident,  $P_a$ , is [57.8] psig. The containment design pressure is 59 psig.
- c. The maximum allowable primary containment leakage rate,  $L_a$ , at  $P_a$ , shall be 0.10% of primary containment air weight per day.
- d. Leakage Rate acceptance criteria are:
  1. Containment leakage rate acceptance criterion is  $1.0 L_a$ . During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are  $\leq 0.60 L_a$  for the Type B and Type C tests and  $\leq 0.75 L_a$  for Type A tests;
  2. Air lock testing acceptance criteria are:
    - a) Overall air lock leakage rate is  $\leq [0.05] L_a$  when tested at  $\geq P_a$ ,
    - b) For each door, leakage rate is  $\leq [0.01] L_a$  when pressurized to  $\geq [10]$  psig.
- e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.
- f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

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## 5.5 Programs and Manuals

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### 5.5.9 System Level OPERABILITY Testing Program

The System Level OPERABILITY Testing Program provides requirements for performance tests of passive systems. The System Level Inservice Tests specified in Section 3.9.6 and Table 3.9-17 apply when specified by individual Surveillance Requirements.

- a. The provisions of SR 3.0.2 are applicable to the test frequencies specified in Table 3.9.17 for performing system level OPERABILITY testing activities; and
- b. The provisions of SR 3.0.3 are applicable to system level OPERABILITY testing activities.

### 5.5.10 Component Cyclic or Transient Limit

This program provides controls to track the Table 3.9-1A cyclic and transient occurrences to ensure that components are maintained within the design limits.

### 5.5.11 Battery Monitoring and Maintenance Program

This Program provides for battery restoration and maintenance, based on [the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," or of the battery manufacturer] including the following:

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- 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 minimum established design limit.
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5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

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

5.6.1 Occupational Radiation Exposure Report

**- NOTE -**

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), electronic dosimeter or film badge measurements. Small exposures totaling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year. [The initial report shall be submitted by April 30 of the year following the initial criticality.]

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5.6.2 Annual Radiological Environmental Operating Report

**- 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.

5.6 Reporting Requirements

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5.6.2 Annual Radiological Environmental Operating Report (continued)

The Annual Radiological Environmental Operating Report shall include the results

## 5.6 Reporting Requirements

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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.

5.6.3 Radioactive Effluent Release Report-----  
- NOTE -  
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A single submittal may be made for a multiple unit station.

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 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

- 2.1.1, "Reactor Core SLs"
- 3.1.1, "SHUTDOWN MARGIN (SDM)"
- 3.1.3, "Moderator Temperature Coefficient"
- 3.1.5, "Shutdown Bank Insertion Limits"
- 3.1.6, "Control Bank Insertion Limits"
- 3.2.1, "Heat Flux Hot Channel Factor"

## BASES

## LCO (continued)

The SLB and the boron dilution accidents (Ref. 2) are the most limiting analyses that establish the SDM value of the LCO. For SLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100 limits (Ref. 3). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for automatic action to terminate dilution may no longer be applicable.

## APPLICABILITY

In MODE 2 with  $k_{\text{eff}} < 1.0$ , and in MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits."

## ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a concentrated solution. The operator should begin boration with the best source available for the plant conditions.

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at **hot shutdown conditions when boron concentration is highest at 1502 ppm**. ~~the beginning of cycle when the boron concentration is highest.~~ Assuming that a value of **1.06%**  $\Delta k/k$  must be recovered and ~~a~~ the boration flow rate is **100** gpm, it is possible to increase the boron concentration of the RCS by **442-111** ppm in approximately **29-21** minutes **utilizing boric acid solution having a concentration of 4375 ppm**. If a boron worth of 9 pcm/ppm is assumed, this combination of parameters will increase the SDM by **1.06%**  $\Delta k/k$ . These boration parameters of **100** gpm and **4375** ppm represent typical values and are provided for the purpose of offering a specific example.

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BASES

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APPLICABLE SAFETY ANALYSES (continued)

LCO 3.1.6 “Control Bank Insertion Limits,” and  
LCO 3.4.2 “Minimum Temperature for Criticality,”

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are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to  $\leq 5\%$  RTP, the reactor coolant temperature is kept  $\geq [535]541^\circ\text{F}$ , and SDM is within the limits provided in the COLR.

PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

Reference 7 allows special test exceptions (STE) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.

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LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

6

- a. RCS lowest loop average temperature is  $\geq [535]541^\circ\text{F}$ ,
- b. SDM is within the limits provided in the COLR, and
- c. THERMAL POWER is  $< 5\%$  RTP.

BASES

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APPLICABILITY      This LCO is applicable when performing low power PHYSICS TESTS. The Applicability is stated as “During PHYSICS TESTS initiated in MODE 2” to ensure that the 5% RPT maximum power level is not exceeded. Should the THERMAL POWER EXCEED 5% RPT, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.

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ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS lowest  $T_{avg}$  is < ~~535~~541°F, the appropriate action is to restore  $T_{avg}$  to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring  $T_{avg}$  to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below ~~535~~541°F could violate the assumptions for accidents analyzed in the safety analyses.

6

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be placed in a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, to reach MODE 3 from MODE 2 HZP conditions in an orderly manner and without challenging plant systems.

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.8.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1 "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS.

SR 3.1.8.2

Verification that the RCS lowest loop  $T_{avg}$  is  $\geq$  ~~535~~541°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will provide assurance that the initial conditions of the safety analyses are not violated.

6

SR 3.1.8.3

Verification that the THERMAL POWER is  $<$  5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

- a. RCS boron concentration;
- b. Control bank position;
- c. RCS average temperature;
- d. Fuel burnup based on gross thermal energy generation;
- e. Xenon concentration;
- f. Samarium concentration; and
- g. Isothermal temperature coefficient (ITC).



BASES

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LCO (continued)

1

The actual values of CFQ are given in the COLR; however, CFQ is normally a number on the order of {2.60}. For the AP1000, the normalized F<sub>Q</sub>(Z) as a function of core height is 1.0.

For RAOC operation, F<sub>Q</sub>(Z) is approximated by F<sub>Q</sub><sup>C</sup>(Z) and F<sub>Q</sub><sup>W</sup>(Z). Thus, both F<sub>Q</sub><sup>C</sup>(Z) and F<sub>Q</sub><sup>W</sup>(Z) must meet the preceding limits on F<sub>Q</sub>(Z).

An F<sub>Q</sub><sup>C</sup>(Z) evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results the measured value of F<sub>Q</sub>(Z), called F<sub>Q</sub><sup>M</sup>(Z) is obtained. Then,

$$F_Q^C(Z) = F_Q^M(Z) * F_Q^{MU}(Z)$$

where F<sub>Q</sub><sup>MU</sup>(Z) is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty. F<sub>Q</sub><sup>MU</sup>(Z) is provided in the COLR..

F<sub>Q</sub><sup>C</sup>(Z) is an excellent approximation for F<sub>Q</sub>(Z) when the reactor is at the steady state power at which the incore flux map was taken.

The expression for F<sub>Q</sub><sup>W</sup>(Z) is:

$$F_Q^W(Z) = F_Q^C(Z) * W(Z)$$

where W(Z) is a cycle-dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR.

The F<sub>Q</sub>(Z) limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA F<sub>Q</sub>(Z) limits. If F<sub>Q</sub>(Z) cannot be maintained within the LCO limits, reduction of the core power is required and if F<sub>Q</sub><sup>W</sup>(Z) cannot be maintained within LCO limits, reduction of the AFD limits will also result in a reduction of the core power.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

required to account for any increase to F<sub>Q</sub><sup>M</sup>(Z) which could occur and cause the F<sub>Q</sub>(Z) limit to be exceeded before the next required F<sub>Q</sub>(Z) evaluation.

1

If the two most recent F<sub>Q</sub>(Z) evaluations show an increase in F<sub>Q</sub><sup>C</sup>(Z), it is required to meet the F<sub>Q</sub>(Z) limit with the last F<sub>Q</sub><sup>W</sup>(Z) increased by **the greater of a factor of {1.02} or by an appropriate factor as specified in the COLR** or to evaluate F<sub>Q</sub>(Z) more frequently, each 7 EFPDs. These alternative requirements will prevent F<sub>Q</sub>(Z) from exceeding its limit for any significant period of time without detection.

Performing the Surveillance in MODE 1 prior to exceeding 75% of RTP ensures that the F<sub>Q</sub>(Z) limit will be met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

The Surveillance Frequency of 31 EFPDs is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with Technical Specifications, to preclude the occurrence of adverse peaking factors between 31 EFPD Surveillances. The Surveillance may be done more frequently if required by the results of F<sub>Q</sub>(Z) evaluations.

F<sub>Q</sub>(Z) is verified at power increases of at least 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions, to assure that F<sub>Q</sub>(Z) will be within its limit at higher power levels.

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REFERENCES

1. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," 1974.
2. Regulatory Guide 1.77, Rev. 0, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.
3. 10 CFR 50, Appendix A, GDC 26.
4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988 (Westinghouse Proprietary) and WCAP-7308-L-A (Non-Proprietary).
5. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification," February 1994 (Westinghouse Proprietary) and WCAP-10217-A (Non-Proprietary).

BASES

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LCO (continued)

The limiting value of  $F_{\Delta H}^N$ , described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of  $F_{\Delta H}^N$  is allowed to increase {0.3}% for every 1% RTP reduction in THERMAL POWER.

1

APPLICABILITY

When the OPDMS is inoperable and core power distribution parameters cannot be continuously monitored, it is necessary to monitor  $F_{\Delta H}^N(Z)$  on a periodic basis. Furthermore,  $F_{\Delta H}^N$  limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and peak cladding temperature (PCT). Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to  $F_{\Delta H}^N$  in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict  $F_{\Delta H}^N$  in these modes.

ACTIONS

A.1.1

With  $F_{\Delta H}^N$  exceeding its limit, the unit is allowed 4 hours to restore  $F_{\Delta H}^N$  to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring  $F_{\Delta H}^N$  within its power-dependent limit.

When the  $F_{\Delta H}^N$  limit is exceeded, it is not likely that the DNBR limit would be violated in steady state operation, since events that could significantly perturb the  $F_{\Delta H}^N$  value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore  $F_{\Delta H}^N$  to within its limits without allowing the plant to remain outside  $F_{\Delta H}^N$  limits for an extended period of time.

BASES

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ACTIONS (continued)

7

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within [6] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [6] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [7].

In addition to placing the inoperable channel(s) in the bypassed or tripped condition, THERMAL POWER must be reduced to  $\leq 75\%$  RTP within 12 hours. Reducing the power level prevents operation of the core with radial power distributions beyond the design limits. With one or two of the PMS power range detectors inoperable, partial radial power distribution monitoring capability is lost. However, the protective function would still function even with a single failure of one of the two remaining channels.

As an alternative to reducing power, the inoperable channel(s) can be placed in the bypassed or tripped condition within 6 hours and the QPTR monitored every 12 hours as per SR 3.2.4.2, QPTR verification. Calculating QPTR compensates for the lost monitoring capability and allows continued plant operation at power levels  $> 75\%$  RTP. The 12 hour Frequency is consistent with LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

Required Action D.2.2 has been modified by a Note which only requires SR 3.2.4.2 to be performed if OPDMS and the Power Range Neutron Flux input to QPTR become inoperable. Power distribution limits are normally verified in accordance with LCO 3.2.5, "OPDMS - Monitored Power Distribution Parameters." However, if OPDMS becomes inoperable, then LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," becomes applicable. Failure of a component in the Power Range Neutron Flux Channel which renders the High Flux Trip Function inoperable may not affect the capability to monitor QPTR. If either OPDMS or the channel input to QPTR is OPERABLE, then performance of SR 3.2.4.2 once per 12 hours is not necessary.

As an alternative to the above Actions, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Twelve hours are allowed to place the plant in MODE 3. This is a

BASES

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ACTIONS (continued)

reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. If Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

E.1.1, E.1.2, and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux – Low;
- Overtemperature  $\Delta T$ ;
- Overpower  $\Delta T$ ;
- Power Range Neutron Flux – High Positive Rate;
- Pressurizer Pressure – High;
- SG Water Level – Low; and
- SG Water Level – High 2.

7

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within [6] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [6] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [7].

If the Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this Function is no longer required to be OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

BASES

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ACTIONS (continued)

F.1.1, F.1.2, F.2, and F.3

Condition F applies to the Intermediate Range Neutron Flux trip when above the P-6 setpoint and below the P-10 setpoint. Above the P-6 setpoint and below the P-10 setpoint, the PMS intermediate range detector performs the monitoring functions.

7

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within [2] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [2] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [7].

As an alternative to placing the channel(s) in bypass or trip if THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 2 hours are allowed to reduce THERMAL POWER below the P-6 setpoint or to increase the THERMAL POWER above the P-10 setpoint. The PMS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the PMS power range detectors perform the monitoring and protective functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment below P-6, and takes into account the redundant capability afforded by the two remaining OPERABLE channels and the low probability of their failure during this period.

G.1 and G.2

Condition G applies to three Intermediate Range Neutron Flux trip channels inoperable in MODE 2 above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the PMS intermediate range detector performs the monitoring Functions. With only one intermediate range channel OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are insufficient

BASES

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ACTIONS (continued)

K.1.1, K.1.2, and K.2

Condition K applies to the following reactor trip Functions:

- Pressurizer Pressure – Low;
- Pressurizer Water Level – High 3;
- Reactor Coolant Flow – Low (Both Hot Legs);
- RCP Bearing Water Temperature – High (Two Pumps); and
- RCP Speed – Low.

7

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within [6] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [6] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [7].

If Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this Function is no longer required to be OPERABLE. A Completion Time of an additional 6 hours is allowed to reduce power < P-10. Allowance of this time interval takes into consideration the redundant capability provided by the remaining two redundant OPERABLE channels and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition K.

L.1.1, L.1.2, and L.2

Condition L is applicable to the Reactor Coolant Flow – Low (Single Cold Leg) and RCP Bearing Water Temperature – High (Single Pump) reactor trip Functions.

7

With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within [6] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the

BASES

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ACTIONS (continued)

7

single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [6] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [7].

If Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this Function is no longer required to be OPERABLE. A Completion Time of an additional 4 hours is allowed to reduce power < P-8. Allowance of this time interval takes into consideration the redundant capability provided by the remaining two redundant OPERABLE channels and the low probability of occurrence of an event during this period that may require the protection afforded by this Function.

M.1 and M.2

Condition M applies to the Safeguards Actuation signal from ESFAS reactor trip, the RTS Automatic Trip Logic, automatic ADS Stages 1, 2, and 3 actuation, and automatic CMT injection in MODES 1 and 2.

With one or two channels or divisions inoperable, the Required Action is to restore three of the four channels/divisions within 6 hours. Restoring all channels/divisions but one to OPERABLE status ensures that a single failure will neither cause nor prevent the protective function. The 6 hour Completion Time is considered reasonable since the protective function will still function.

If Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this Function is no longer required to be OPERABLE. A Completion Time of an additional 6 hours is allowed to place the unit in MODE 3. The Completion Time is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. Allowance of this time interval takes into consideration the redundant capability provided by the remaining two redundant OPERABLE channels/divisions and the low probability of occurrence of an event during this period that may require the protection afforded by this Function.



BASES

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ACTIONS (continued)

N.1, N.2.1, N.2.2, and N.3

7

Condition N applies to the P-6, P-10, and P-11 interlocks. With one or two channels inoperable, the associated interlock must be verified to be in its required state for the existing plant condition within 1 hour, or the Functions associated with inoperable interlocks placed in a bypassed or tripped condition within [7] hours, or the unit must be placed in MODE 3 within 13 hours. Verifying the interlock manually accomplishes the interlock condition.

7

If one interlock channel is inoperable, the associated Function(s) must be placed in a bypass or trip condition within [7] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.)

7

If two interlock channels are inoperable, one channel of the associated Function(s) must be bypassed and one channel of the associated Function(s) must be tripped. In this state, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [7] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [7].

If placing the associated Functions in bypass or trip is impractical, for instance as the result of other channels in bypass or trip, the Completion Time of an additional 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

7

O.1, O.2.1, O.2.2, and O.3

Condition O applies to the P-8 interlock. With one or two channels inoperable, the associated interlock must be verified to be in its required state for the existing plant condition within 1 hour, or the Functions associated with inoperable interlocks placed in a bypassed or tripped condition within [7] hours, or the unit must be placed in MODE 2 within 13 hours. Verifying the interlock manually accomplishes the interlock condition.

BASES

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ACTIONS (continued)

7

If one interlock channel is inoperable, the associated Function(s) must be placed in a bypass or trip condition within [7] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.)

7

If two interlock channels are inoperable, one channel of the associated Function(s) must be bypassed and one channel of the associated Function(s) must be tripped. In this state, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [7] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [7].

If placing the associated Functions in bypass or trip is impractical, for instance as the result of other channels in bypass or trip, the Completion Time of an additional 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging plant systems.

P.1, P.2.1, and P.2.2

Condition P applies to the RTBs, and RTB undervoltage and shunt trip mechanisms in MODES 1 and 2, and in MODES 3, 4, and 5 with the RTBs closed and the PLS capable of rod withdrawal. This Condition is primarily associated with mechanical damage that can prevent the RTBs from opening.

With one division inoperable, the reactor trip breakers in the inoperable division must be opened within 8 hours. A division is inoperable, if, within that division, one or both of the RTBs and/or one or both of the trip mechanisms is inoperable.

With one division inoperable (with its RTBs open) and with three OPERABLE divisions remaining, the trip logic becomes one-out-of-three. The one-out-of-three trip logic meets the single failure criterion. (A failure in one of the three remaining divisions will not prevent the protective function.) If, coincident with RTBs inoperable in one division, the automatic trip logic is inoperable in another division, the trip logic becomes one-out-of-two, which meets the single failure criterion.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

A Note modifies SR 3.3.1.4. The Note states that this Surveillance is required only if reactor power is > 50% RTP and that 24 hours is allowed for performing the first surveillance after reaching 50% RTP.

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

SR 3.3.1.5

SR 3.3.1.5 is the performance of a TADOT every 92 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

The Reactor Trip Breaker (RTB) test shall include separate verification of the undervoltage and shunt trip mechanisms. Each RTB in a division shall be tested separately in order to minimize the possibility of an inadvertent trip.

The Frequency of every 92 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data. In addition, the AP1000 design provides additional breakers to enhance reliability.

The SR is modified by a Note to clarify that both breakers in a single division are to be tested during each STAGGERED TEST.

SR 3.3.1.6

7

SR 3.3.1.6 is the performance of a REACTOR TRIP CHANNEL OPERATIONAL TEST (RTCOT) every [92] days.

A RTCOT is performed on each required channel to provide reasonable assurance that the entire channel will perform the intended Function.

A test subsystem is provided with the protection and safety monitoring system to aid the plant staff in performing the RTCOT. The test subsystem is designed to allow for complete functional testing by using a combination of system self checking features, functional testing features, and other testing features. Successful functional testing consists of verifying that the capability of the system to perform the safety function has not failed or degraded.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

For hardware functions this would involve verifying that the hardware components and connections have not failed or degraded. Generally this verification includes a comparison of the outputs from two or more redundant subsystems or channels.

Since software does not degrade, software functional testing involves verifying that the software code has not changed and that the software code is executing.

To the extent possible, protection and safety monitoring system functional testing is accomplished with continuous system self-checking features and the continuous functional testing features. The RTCOT shall include a review of the operation of the test subsystem to verify the completeness and adequacy of the results.

If the RTCOT can not be completed using the built-in test subsystem, either because of failures in the test subsystem or failures in redundant channel hardware used for functional testing, the RTCOT can be performed using portable test equipment.

This test frequency of [92] days is justified based on Reference [7] and the use of continuous diagnostic test features, such as deadman timers, cross-check of redundant channels, memory checks, numeric coprocessor checks, and tests of timers, counters and crystal time bases, which will report a failure within the protection and safety monitoring system cabinets to the operator within 10 minutes of a detectable failure.

7

SR 3.3.1.6 is modified by a note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.6 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for a time greater than 4 hours, this Surveillance must be performed prior to 4 hours after entry into MODE 3.

During the RTCOT, the protection and safety monitoring system cabinets in the division under test may be placed in bypass.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a RTCOT as described in SR 3.3.1.6, except it is modified by a Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit

BASES

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SURVEILLANCE REQUIREMENTS (continued)

measure response times. Experience has shown that these components usually pass this surveillance when performed on a refueling frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR 3.3.1.11 is modified by exempting neutron detectors from response time testing. A Note to the Surveillance indicates that neutron detectors may be excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

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REFERENCES

1. Chapter 6.0, "Engineered Safety Features."
2. Chapter 7.0, "Instrumentation and Controls."
3. Chapter 15.0, "Accident Analysis."
4. WCAP-14606, "Westinghouse Setpoint Methodology for Protection Systems," April 1996 (nonproprietary).
5. Institute of Electrical and Electronic Engineers, IEEE-603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," June 27, 1991.
6. 10 CFR 50.49, "Environmental Qualifications of Electric Equipment Important to Safety for Nuclear Power Plants."
7. WCAP-10271-P-A (Proprietary) and WCAP-10272-A (Non-Proprietary), "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," Supplement 2, Revision 1, June 1990.]
8. NRC Generic Letter No. 83-27, Surveillance Intervals in Standard Technical Specifications.
9. ESBU-TB-97-01, Westinghouse Technical Bulletin, "Digital Process Rack Operability Determination Criteria," May 1, 1997.
10. WCAP-13632-P-A (Proprietary) and WCAP-13787-A (Non-Proprietary), Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.

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BASES

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

operator can manually block low pressurizer level signal used for these actuations. Concurrent with blocking CMT actuation on low pressurizer level, IRWST actuation on Low 2 RCS hot leg level is enabled. When the pressurizer level is above the P-12 setpoint, the pressurizer level signal is automatically enabled and a confirmatory open signal is issued to the isolation valves on the CMT cold leg balance lines. This Function is required to be OPERABLE in MODES 1, 2, 3, 4, 5, and 6.

18.e. RCS Pressure, P-19

10

The P-19 interlock is provided to permit water solid conditions (i.e., when the pressurizer water level is >[92]%) in lower MODES without automatic isolation of the CVS makeup pumps. With RCS pressure below the P-19 setpoint, the operator can manually block CVS isolation on High 2 pressurizer water level. When RCS pressure is above the P-19 setpoint, this Function is automatically unblocked. This Function is required to be OPERABLE IN MODES 1, 2, 3, and 4 with the RCS not being cooled by the RNS. When the RNS is cooled by the RNS, the RNS suction relief valve provides the required overpressure protection (LCO 3.4.14).

19. Containment Air Filtration System Isolation

Some DBAs such as a LOCA may release radioactivity into the containment where the potential would exist for the radioactivity to be released to the atmosphere and exceed the acceptable site dose limits. Isolation of the Containment Air Filtration System provides protection to prevent radioactivity inside containment from being released to the atmosphere.

19.a. Containment Radioactivity – High 1

Three channels of Containment Radioactivity – High 1 are required to be OPERABLE in MODES 1, 2, 3, and 4 with the RCS not being cooled by the RNS, when the potential exists for a LOCA, to protect against radioactivity inside containment being released to the atmosphere. These Functions are not required to be OPERABLE in MODE 4 with the RCS being cooled by the RNS or MODES 5 and 6, because any DBA release of radioactivity into the containment in these MODES would not require containment isolation.

BASES

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ACTIONS (continued)

to refer to Table 3.3.2-1 and to take the Required Actions for the protection Functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1 and B.2

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With one or two channels or divisions inoperable, one affected channel or division must be placed in a bypass or trip condition within [6] hours. If one channel or division is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels or divisions will not prevent the protective function.) If one channel or division is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels or divisions will not prevent the protective function.) If one channel or division is bypassed and one channel or division is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [6] hours allowed to place the inoperable channel(s) or division(s) in the bypassed or tripped condition is justified in Reference [6].

C.1

7

With one channel inoperable, the affected channel must be placed in a bypass condition within [6] hours. The [6] hours allowed to place the inoperable channel in the bypass condition is justified in Reference [6]. If one CVS isolation channel is bypassed, the logic becomes one-out-of-one. A single failure in the remaining channel could cause a spurious CVS isolation. Spurious CVS isolation, while undesirable, would not cause an upset plant condition.

D.1

With one required division inoperable, the affected division must be restored to OPERABLE status within 6 hours.

Condition D applies to one inoperable required division of P-4 Interlock (Function 18.a). With one required division inoperable, the 2 remaining OPERABLE divisions are capable of providing the required interlock function, but without a single failure. The P-4 Interlock is enabled when RTBs in two divisions are detected as open. The status of the other inoperable, non-required P-4 division is not significant, since P-4 divisions can not be tripped or bypassed. In order to provide single failure tolerance, 3 required divisions must be OPERABLE.

BASES

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ACTIONS (continued)

inoperable, the inoperable channel must be placed in a trip condition within 6 hours. If one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The specified Completion Time is reasonable considering the time required to complete this action.

I.1 and I.2

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Condition I applies to IRWST containment recirculation valve actuation on safeguards actuation coincident with IRWST Level Low 3 (Function 23.b). With one or two channels inoperable, one affected channel must be placed in a bypass or trip condition within [6] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is bypassed and one channel is tripped, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [6] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [6].

J.1 and J.2

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Condition J applies to the P-6, P-11, P-12, and P-19 interlocks. With one or two required channel(s) inoperable, the associated interlock must be verified to be in its required state for the existing plant condition within 1 hour, or any Function channels associated with inoperable interlocks placed in a bypassed condition within [7] hours. Verifying the interlock state manually accomplishes the interlock role.

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If one interlock channel is inoperable, the associated Function(s) must be placed in a bypass or trip condition within [7] hours. If one channel is bypassed, the logic becomes two-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.) If one channel is tripped, the logic becomes one-out-of-three, while still meeting the single failure criterion. (A failure in one of the three remaining channels will not prevent the protective function.)

If two interlock channels are inoperable, one channel of the associated Function(s) must be bypassed and one channel of the associated



BASES

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ACTIONS (continued)

7

Function(s) must be tripped. In this state, the logic becomes one-out-of-two, while still meeting the single failure criterion. The [7] hours allowed to place the inoperable channel(s) in the bypassed or tripped condition is justified in Reference [6].

K.1

LCO 3.08 is applicable while in MODE 5 or 6. Since irradiated fuel assembly movement can occur in MODE 5 or 6, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, the fuel movement is independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

Condition K is applicable to the MCR Isolation and Air Supply Initiation (Function 20), during movement of irradiated fuel assemblies. If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must suspend movement of the irradiated fuel assemblies immediately. The required action suspends activities with potential for releasing radioactivity that might enter the MCR. This action does not preclude the movement of fuel to a safe position.

L.1

If the required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a MODE in which the LCO does not apply. This accomplished by placing the plant in MODE 3 within 6 hours. The allowed time is reasonable, based operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

M.1 and M.2

If the Required Action and associated Completion Time of the first condition listed in Table 3.3.2-1 is not met, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Frequency of every 92 days on a STAGGERED TEST BASIS provides a complete test of all four divisions once per year. This frequency is adequate based on the inherent high reliability of the solid state devices which comprise this equipment; the additional reliability provided by the redundant subsystems; and the use of continuous diagnostic test features, such as deadman timers, memory checks, numeric coprocessor checks, cross-check of redundant subsystems, and tests of timers, counters, and crystal time basis, which will report a failure within these cabinets to the operator.

SR 3.3.2.3

SR 3.3.2.3 is the performance of a TADOT of the manual actuations, initiations, and blocks for various ESF Functions, the Class 1E battery charger undervoltage inputs, and the reactor trip (P-4) input from the IPCs. This TADOT is performed every 24 months.

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

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The setpoints for the Class 1E battery charger undervoltage relays require bench calibration and are verified during CHANNEL CALIBRATION. The other functions have no setpoints associated with them.

SR 3.3.2.4

SR 3.3.2.4 is the performance of a CHANNEL CALIBRATION every 24 months or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor and the IPC.

The Frequency is based on operating experience and consistency with the refueling cycle.

This Surveillance Requirement is modified by a Note. The Note states that this test should include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.2.5

SR 3.3.2.5 is the performance of an CHANNEL OPERATIONAL TEST (COT) every [92] days.

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

A COT is performed on each required channel to provide reasonable assurance that the entire channel will perform the intended ESF Function.

A test subsystem is provided with the protection and safety monitoring system to aid the plant staff in performing the COT. The test subsystem is designed to allow for complete functional testing by using a combination of system self-checking features, functional testing features, and other testing features. Successful functional testing consists of verifying that the capability of the system to perform the safety function has not failed or degraded.

For hardware functions this would involve verifying that the hardware components and connections have not failed or degraded. Generally this verification includes a comparison of the outputs from two or more redundant subsystems or channels.

Since software does not degrade, software functional testing involves verifying that the software code has not changed and that the software code is executing.

To the extent possible, protection and safety monitoring system functional testing is accomplished with continuous system self-checking features and the continuous functional testing features. The COT shall include a review of the operation of the test subsystem to verify the completeness and adequacy of the results.

If the COT can not be completed using the built-in test subsystem, either because of failures in the test subsystem or failures in redundant channel hardware used for functional testing, the COT can be performed using portable test equipment.

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The [92]-day Frequency is based on Reference 6 and the use of continuous diagnostic test features, such as deadman timers, cross-check of redundant channels, memory checks, numeric coprocessor checks, and tests of timers, counters and crystal time bases, which will report a failure within the integrated protection cabinets to the operator.

During the COT, the protection and safety monitoring system cabinets in the division under test may be placed in bypass.

SR 3.3.2.6

This SR ensures the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis.

BASES

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SURVEILLANCE REQUIREMENTS (continued)

The Frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation. This Frequency is adequate based on the use of multiple circuit breakers to prevent the failure of any single circuit breaker from disabling the function and that all circuit breakers are tested.

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REFERENCES

1. Chapter 6, "Engineered Safety Features."
2. Chapter 7, "Instrumentation and Controls."
3. Chapter 15, "Accident Analysis."
4. Institute of Electrical and Electronic Engineers, IEEE-603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," June 27, 1991.
5. 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."
6. WCAP-10271-P-A (Proprietary) and WCAP-10272-A (Non-Proprietary), Supplement 2, Rev. 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," dated June 1990.]
7. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants."
8. NUREG-1218, "Regulatory Analysis for Resolution of USI A-47," 4/88.
9. WCAP-14606, "Westinghouse Setpoint Methodology for Protection Systems," April 1996 (nonproprietary).
10. ESBU-TB-97-01, Westinghouse Technical Bulletin, "Digital Process Rack Operability Determination Criteria," May 1, 1997.
11. WCAP-13632-P-A (Proprietary) and WCAP-13787-A (Non-Proprietary), Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.

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BASES

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APPLICABLE SAFETY ANALYSES (continued)

PAM Instrumentation that is required in accordance with Regulatory Guide 1.97 satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

The PAM instrumentation LCO provides OPERABILITY requirements for those monitors which provide information required by the control room operators to assess the process of accomplishing or maintaining critical safety functions. This LCO addresses those Regulatory Guide 1.97 instruments which are listed in Table 3.3.3-1.

The OPERABILITY of the PAM Instrumentation ensures there is sufficient information available on selected plant parameters to monitor and assess plant status following an accident. This capability is consistent with the recommendations of Reference 1.

Category 1 non-type A variables are required to meet Regulatory Guide 1.97 Category 1 (Ref. 1) design and qualification requirements for seismic and environmental qualification, single-failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display.

Listed below are discussions of the specified instrument functions listed in Table 3.3.3-1. Each of these is a Category 1 variable.

1. Intermediate Range Neutron Flux

Neutron Flux indication is provided to verify reactor shutdown. The neutron flux intermediate range is sufficient to cover the full range of flux that may occur post accident.

Neutron flux is used for accident diagnosis, verification of subcriticality, and diagnosis of positive reactivity insertion.

2, 3. Reactor Coolant System (RCS) Wide Range Hot and Cold Leg Temperature

RCS Hot and Cold Leg Temperatures are provided for verification of core cooling and long-term surveillance. The channels provide indication over a range of {50}°F to {700}°F.

In addition to this, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify the plant conditions necessary to establish natural circulation in the RCS.

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BASES

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LCO (continued)

4. RCS Pressure

RCS wide range pressure is provided for verification of core cooling and RCS integrity long term surveillance.

5. Pressurizer Pressure and RCS Subcooling Monitor

Pressurizer Pressure is used to determine RCS Subcooling. The RCS Subcooling Monitor is provided for verification of core cooling. Subcooling margin is available when the RCS pressure is greater than the saturation pressure corresponding to the core exit temperature. Inputs to the Subcooling Monitor are pressurizer pressure and RCS hot leg temperature.

6. Containment Water Level

Containment Water Level is used to monitor the containment environment during accident conditions. The containment water level can also provide information to the operators that the various stages of safety injection along with system depressurization are progressing.

7. Containment Pressure

The containment pressure transmitters monitor the containment pressure over the range of [-5] to [10] psig. This provides information on post accident containment pressure and containment integrity.

8. Containment Pressure (Extended Range)

The extended range containment pressure transmitters are instruments that operators use for monitoring the potential for breach of containment, a fission product barrier. The extended range sensors monitor containment pressure over the range of [0] to [240] psig.

9. Containment Area Radiation (High Range)

Containment Area Radiation is provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

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BASES

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APPLICABLE SAFETY ANALYSES (continued)

dropped or stuck rod events. An assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit and the RCS average temperature limit specified in the COLR correspond to analytical limits, with an allowance for steady state fluctuations and measurement errors. The RCS average temperature limit corresponds to the analytical limit with allowance for controller deadband and measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

2

This LCO specifies limits on the monitored process variables, pressurizer pressure, RCS average temperature, and RCS total flow rate to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, usually based on [maximum analyzed steam generator tube plugging], is retained in the TS LCO. Operating within these limits will result in meeting DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error based on performing a precision heat balance and using the result to normalize the RCS flow rate indicators.

The numerical values for pressure, temperature, and flow rate specified in the COLR are given for the measurement location but have been adjusted for instrument error.

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APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state plant operation in order to ensure DNBR criterion will be met in the event of an unplanned loss of forced coolant flow or other DNB-limiting transient. In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.2.1

6

RCS loop average temperature is required to be verified at or above {551}°F every 12 hours. The SR to verify RCS loop average temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.

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REFERENCES

1. Chapter 15, "Accident Analyses."
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BASES

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APPLICABLE SAFETY ANALYSES (continued)

Therefore, in MODE 3, 4 or 5 with the RTBs in the closed position and the PLS capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires the RCPs to be OPERABLE and in operation to ensure that the accident analysis limits are met.

In MODES 3, 4 and 5 with the RTBs open, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. This is addressed in LCO 3.4.8, "Minimum RCS Flow."

RCS Loops satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required in MODES 1 and 2. The requirement that at least four RCPs must be operating in MODES 3, 4 and 5 when the RTBs are closed provides assurance that, in the event of a rod withdrawal accident, there will be adequate flow in the core to avoid exceeding the DNBR limit. Bypass of the RCP variable speed control ensures that the pumps are operating at full flow.

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With the RTBs in the open position, the PLS is not capable of rod withdrawal; therefore only a minimum RCS flow of 103,000 gpm is necessary to ensure removal of decay heat from the core in accordance with LCO 3.4.8, Minimum RCS Flow.

Note 1 prohibits startup of a RCP when the reactor trip breakers are closed. This requirement prevents startup of a RCP and the resulting circulation of cold and/or unborated water from an inactive loop into the core, precluding reactivity excursion events which are unanalyzed.

**Note 2 prohibits startup of an RCP when the RCS temperature is  $\geq 200^{\circ}\text{F}$  unless pressurizer level is  $< 92\%$ . This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.**

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Note 32 requires that the secondary side water temperature of each SG be  $\leq \{50\}^{\circ}\text{F}$  above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature  $\leq 20075^{\circ}\text{F}$ . This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

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BASES

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LCO (continued)

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Note 43 permits all RCPS to be de-energized in MODE 3, 4, or 5 for  $\leq 1$  hour per 8 hour period. The purpose of the NOTE is to permit tests that are designed to validate various accident analysis values. One of these tests is for the validation of the pump coastdown curve, used as input to a number of accident analyses including a loss of flow accident.

~~LCO (continued)~~

This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve may need to be revalidated by conducting the test again.

Another test performed during the startup testing program is the validation of the rod drop times during cold conditions, both with and without flow.

The no-flow tests may be performed in MODE 3, 4, or 5, and require that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should only be performed once, unless the flow characteristics of the RCS are changed. The 1 hour time period specified is adequate to perform the desired tests and experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of the NOTE is permitted provided the following conditions are met along with any other conditions imposed by initial startup test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause natural circulation flow obstruction.

An OPERABLE RCS loop is composed of two OPERABLE RCPs in operation providing forced flow for heat transport and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program,

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.6.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested one at a time and in accordance with the requirements of ASME Code Section XI (Ref. 4), which provides the activities and Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is  $\pm 3\%$  for OPERABILITY; however, the values are reset to  $\pm 1\%$  during the Surveillance to allow for drift.

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REFERENCES

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1. ASME Boiler and Pressure Vessel Code, Section III, NB 7614.3.
  2. [WCAP-7769, "Topical Report on Overpressure Protection, October 1971."]
  3. Chapter 15, "Accident Analyses."
  4. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."
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BASES

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LCO (continued)

NOTE 1 permits all RCPS to be de-energized for  $\leq 1$  hour per 8 hour period. The purpose of the NOTE is to permit tests that are designed to validate various accident analysis values. One of these tests is for the validation of the pump coastdown curve, used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve may need to be revalidated by conducting the test again.

Another test performed during the startup testing program is the validation of the rod drop times during cold conditions, both with and without flow.

The no-flow tests may be performed in MODE 3, 4, or 5, and require that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should only be performed once, unless the flow characteristics of the RCS are changed. The 1 hour time period specified is adequate to perform the desired tests and experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of the NOTE is permitted provided the following conditions are met along with any other conditions imposed by initial startup test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause natural circulation flow obstruction.

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**Note 2 prohibits startup of an RCP when the RCS temperature is  $\geq 200^\circ\text{F}$  unless pressurizer level is  $< 92\%$ . This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.**

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Note 23 requires that the secondary side water temperature of each SG be  $\leq [50]^\circ\text{F}$  above each of the RCS cold leg temperatures before the start of

BASES

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LCO (continued)

1

an RCP with any RCS cold leg temperature  $\leq 20075^{\circ}\text{F}$ . This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

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APPLICABILITY

Minimum RCS flow is required in MODES 3, 4, and 5 with the reactor trip breakers (RTBs) open because an inadvertent BDE is considered possible in these MODES.

In MODES 1 and 2, and in MODES 3, 4, and 5 with the RTBs closed, LCO 3.4.4 requires all four RCPs to be in operation. Thus, in the event of an inadvertent boron dilution, adequate mixing will occur.

A minimum mixing flow is not required in MODE 6 because LCO 3.9.2 requires that all valves used to isolate unborated water sources shall be secured in the closed position. In this situation, an inadvertent BDE is not considered credible.

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ACTIONS

A.1

If no RCP is in operation, all sources of unborated water must be isolated within 1 hour. This action assures that no unborated water will be introduced into the RCS when proper mixing cannot be assured. The allowed Completion Time requires that prompt action be taken, and is based on the low probability of a DBA occurring during this time.

A.2

The Requirement to perform SR 3.1.1.1 (SDM verification) within 1 hour assures that if the boron concentration in the RCS has been reduced and not detected by the source range instrumentation, prompt action may be taken to restore the required SDM. The allowed Completion Time is consistent with that required of Action A.1 because the conditions and consequences are the same.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.8.1

This Surveillance requires verification every 12 hours that a minimum mixing flow is present in the RCS. A Frequency of 12 hours is adequate considering the low probability of an inadvertent BDE during this time, and the ease of verifying the required RCS flow.

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BASES

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BACKGROUND (continued)

ADS stages 1, 2 and 3 valves are designed to open relatively slowly, from approximately 25 seconds for the first stage valves, to approximately 70 seconds for the second and third stage valves.

The ADS valves are powered by batteries. In the unlikely event that offsite and onsite AC power is lost for an extended period of time, a timer will actuate ADS within 24 hours of the time at which AC power is lost, before battery power has been degraded to the point where the valves cannot be opened.

The number and capacity of the ADS flow paths are selected so that adequate safety injection is provided from the accumulators, IRWST and containment recirculation for the limiting DBA loss of coolant accident (LOCA). For small break LOCAs the limiting single failure is the loss of one fourth stage flow path (Ref. 2). The PRA (Ref. 3) shows that adequate core cooling can be provided with the failure of up to [seven] (all ADS stage 1 to 3 and [one] ADS stage 4) flow paths. The ADS PRA success criteria following a LOCA or non-LOCA with failure of other decay heat removal features is for 3 of 4 ADS stage 4 valves to open. All of the ADS stage 1, 2, 3 valves can fail to open. This ADS capacity is sufficient to support PXS gravity injection and containment recirculation operation.

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APPLICABLE  
SAFETY  
ANALYSES

For non-LOCA events, use of the ADS is not required and is not anticipated. For these events, injection of borated water into the core from the CMTs may be required for makeup or boration. However, the amount of water necessary will not reduce the level in the CMTs to the point of ADS actuation.

For events which involve a loss of primary coolant inventory, such as a LOCA, the ADS will be actuated, allowing for injection from the accumulators, the IRWST, and the containment recirculation (Ref. 2).

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

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LCO

The requirement that the 16 ADS valves be OPERABLE ensures that upon actuation, the depressurization of the RCS will proceed smoothly and completely, as assumed in the DBA safety analyses.

For the ADS to be considered OPERABLE, the 16 ADS valves must be closed and OPERABLE (capable of opening on an actuation signal). In addition, the stage 4 motor operated isolation valves must be open. These stage 4 motor operated isolation valves are not required to be OPERABLE because they are maintained open per SR 3.4.11.1.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

RCS Vent Performance

13

With the RCS depressurized, a vent size of [9.3] square inches is capable of mitigating a limiting overpressure transient. The area of the vent is equivalent to the area of the inlet pipe to the RNS suction relief valve so the capacity of the vent is greater than the flow possible with either the mass or heat input transient, while maintaining the RCS pressure less than the minimum of either the maximum pressure on the P/T limit curve or 110 percent of the design pressure of the normal residual heat removal system.

The required vent area may be obtained by opening one ADS Stage 2, 3, or 4 flow path.

The RCS vent size will be reevaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the maximum coolant input and minimum pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires all accumulator discharge isolation valves closed and immobilized, when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed in the PTLR.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

- a. One OPERABLE RNS suction relief valve; or

An RNS suction relief valve is OPERABLE for LTOP when both RNS suction isolation valves in one flow path are open, its setpoint is within limits, and testing has proven its ability to open at this setpoint.

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BASES

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LCO (continued)

13

- b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of  $\geq [9.3]$  square inches.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

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APPLICABILITY

This LCO is applicable in MODE 4 when any cold leg temperature is below 275°F, MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above 275°F. In MODE 6, the reactor vessel head is off, and overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.6, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 with the RNS isolated or RCS temperature  $\geq 275^\circ\text{F}$ .

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure with little or no time for operator action to mitigate the event.

The Applicability is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves.

This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

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ACTIONS

A.1, B.1, and B.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action B.1 and Required Action B.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.14.2

The RNS suction relief valve shall be demonstrated OPERABLE by verifying two RNS suction isolation valves in one flow path are open. This Surveillance is only performed if the RNS suction relief valve is being used to satisfy this LCO.

The RNS suction isolation valves are verified to be opened every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RNS suction isolation valves remain open.

SR 3.4.14.3

13

The RCS vent of  $\geq [9.3]$  square inches is proven OPERABLE by verifying its open condition either:

- a. Once every 12 hours for a valve that is not locked (valves that are sealed or secured in the open position are considered "locked" in this context) or
- b. Once every 31 days for other vent path(s) (e.g., a vent valve that is locked, sealed, or secured in position or a removed pressurizer safety valve or open manway also fits this category).

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.14b.

SR 3.4.14.4

The RNS suction relief valve shall be demonstrated OPERABLE by verifying that two RNS suction isolation valves in one flow path are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.14.2 for the RNS suction isolation valve Surveillance.) This Surveillance is only required to be performed if the RNS suction relief valve is being used to meet this LCO. The ASME Code, Section XI (Ref. 5), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.3 Containment Isolation Valves

#### BASES

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#### BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Section 6.2 (Ref. 1) identifies parameters which initiate isolation signal generation for containment isolation valves. The containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analysis. Therefore, the OPERABILITY requirements provide assurance that containment function assumed in the safety analysis will be maintained.

#### Containment Air Filtration System [16-inch] purge valves

1

The Containment Air Filtration System operates to:

- a. Supply outside air into the containment for ventilation and cooling or heating,
- b. Reduce the concentration of noble gases within containment prior to and during personnel access, and
- c. Equalize internal and external pressures.

BASES

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ACTIONS (continued)

acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

D.1 and D.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.3.1

1 This SR ensures that the {16 inch} purge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the {16 inch} purge valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The {16 inch} purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.2.

SR 3.6.3.2

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for valves outside containment is relatively easy, the 31 day Frequency is based on

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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.4 Containment Pressure

#### BASES

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**BACKGROUND** The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of transients which result in a negative pressure.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the operating band of conditions used in the containment pressure analyses for the Design Basis Events which result in internal or external pressure loads on the containment vessel. Should operation occur outside these limits, the initial containment pressure would be outside the range used for containment pressure analyses.

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**APPLICABLE  
SAFETY  
ANALYSES**

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients (Ref. 1).

The initial pressure condition used in the containment analysis was 15.7 psia (1.0 psig). This resulted in a maximum peak pressure from a LOCA,  $P_a$ , of {57.8} psig. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure results from the SLB. The maximum containment pressure resulting from the SLB, {57.3} psig, does not exceed the containment design pressure, 59 psig.

The containment was also designed for an external pressure load equivalent to 2.9 psig. The limiting negative pressure transient is a loss of all AC power sources coincident with extreme cold weather conditions which cool the external surface of the containment vessel. The initial pressure condition used in this analysis was -0.2 psig. This resulted in a minimum pressure inside containment, as illustrated in Reference 1, which is less than the design load. Other external pressure load events evaluated include:

Failed fan cooler control

Malfunction of containment purge system

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BASES

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BACKGROUND (continued)

following an event. The need to close containment for the mid-loop period following a refueling must be evaluated since decay heat varies with the time after shutdown and the impact of the partial core replacement with new fuel. It is expected that containment will be closed for activities where drain-down is planned, such as the RCS drain-down from no-load pressurizer level for the initial mid-loop period during a refueling. Containment is not expected to be closed for minor, unplanned RCS volume transients, such as a short-term inventory where the pressurizer level may be reduced, but not emptied, and where recovery actions are within the time to containment closure.

The containment equipment hatches, which are part of the containment pressure boundary, provide a means for moving large equipment and components into and out of containment. If closed, the equipment hatch must be held in place by at least {four} bolts. Good engineering practice dictates that bolts required by this LCO be approximately equally spaced. Alternatively, if open, each equipment hatch can be installed using a dedicated set of hardware, tools and equipment. A self-contained power source is provided to drive each hoist while lowering the hatch into position. Large equipment and components may be moved through the hatches as long as they can be removed and the hatch closed prior to steaming into the containment.

7

~~Reviewers Note:~~ The design of the equipment hatch is such that the {four} bolts would only be needed to support the hatch in place and provide adequate strength to support the hatch dead weight and associated loads. The hatch is installed on the inside containment and is held in place against a matching flange surface with mating bolt pattern by the bolts. Once the dead weight is supported, any pressure (greater than atmospheric) within containment will serve to exert closure force on the hatch toward the mating flange surface serving to reduce stresses on bolts. Therefore the determination of the number of bolts is limited to the quantity required to support the hatch itself and not related to any potential containment pressure.

7

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for

BASES

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| LCO

7

The requirement to maintain the pH adjustment baskets with  $\geq$  {560} ft<sup>3</sup> of TSP assures that for DBA releases of iodine into containment, the pH of the containment sump will be adjusted to enhance the retention of the iodine.

A required volume is specified instead of mass because it is not feasible to weigh the TSP in the containment. The minimum required volume is based on the manufactured density of TSP. This is conservative because the density of TSP may increase after installation due to compaction.

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APPLICABILITY

In MODES 1, 2, 3, and 4 a DBA could cause release of radioactive iodine to containment requiring pH adjustment. The pH adjustment baskets assist in reducing the airborne iodine fission product inventory available for release to the environment.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, pH adjustment is not required to be OPERABLE in MODES 5 and 6.

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ACTIONS

A.1

If the TSP volume in the baskets is not within limits, the iodine retention may be less than that assumed in the accident analysis for the limiting DBA. Due to the very low probability that the volume of TSP may change, the variations are expected to be minor such that the required capability is substantially available. The 72 hour Completion Time for restoration to within limits is consistent with times applied to minor degradations of ECCS parameters.

B.1 and B.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.9.1

7

The minimum amount of TSP is ~~{560}~~ ft<sup>3</sup>. A volume is specified since it is not feasible to weigh the TSP contained in the pH adjustment baskets. This volume is based on providing sufficient TSP to buffer the post accident containment water to a minimum pH of 7.0. Additionally, the TSP volume is based on treating the maximum volume of post accident water (~~{918,600}~~**908,000** gallons) containing the maximum amount of boron (~~{3009}~~**2990** ppm) as well as other sources of acid. The minimum required mass of TSP is ~~{27,540}~~**26,460** pounds.

The minimum required volume of TSP is based on this minimum required mass of TSP, the minimum density of TSP plus margin to account for degradation of TSP during plant operation. The minimum TSP density is based on the manufactured density, since the density may increase and the volume decrease, during plant operation, due to agglomeration from humidity inside the containment. The minimum required TSP volume also has about 10% margin to account for degradation of TSP during plant operation.

The periodic verification is required every 24 months, since access to the TSP baskets is only feasible during outages, and normal fuel cycles are scheduled for 24 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of TSP placed in the containment building.

SR 3.6.9.2

7

Testing must be performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. A representative sample of ~~{2-36}~~**2.39** grams of TSP from one of the baskets in containment is submerged in  $\geq 1$  liter of water at a boron concentration of ~~{3009}~~**2990** ppm and at the standard temperature of  $25 \pm 5^\circ\text{C}$ . Without agitation, the solution pH should be raised to  $\geq 7.0$  within 4 hours.

7

The minimum required amount of TSP is sufficient to buffer the maximum amount of boron ~~{3009}~~**2990** ppm, the maximum amount of other acids, and the maximum amount of water ~~{918,600}~~**908,000** gallons that can exist in the containment following an accident and achieve a minimum pH of 7.0.

Agitation of the test solution is prohibited, since an adequate standard for the agitation intensity cannot be specified. The test time of 4 hours is necessary to allow time for the dissolved TSP to naturally diffuse through the sample solution. In the post LOCA sump area, rapid mixing would occur due to liquid flow, significantly decreasing the actual amount of time

## BASES

## APPLICABLE SAFETY ANALYSES (continued)

All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the Overtemperature  $\Delta T$  or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the atmospheric or condenser steam dump valves. The DCD Section [15.4.2] safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO.

1

The DCD safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization may be determined by system transient analyses or conservatively arrived at by a simple heat balance calculation. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Power Range Neutron Flux-High reactor trip function. For example, if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrawal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value.

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.



BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.6.1

The MCR air temperature is checked at a frequency of 24 hours to verify that the VBS is performing as required to maintain the initial condition temperature assumed in the safety analysis, and to ensure that the MCR temperature will not exceed the required conditions after loss of VBS cooling. The surveillance limit of 75°F is the initial heat sink temperature assumed in the VES thermal analysis. The 24 hour Frequency is acceptable based on the availability of temperature indication in the MCR.

SR 3.7.6.2

Verification every 24 hours that compressed air storage tanks are pressurized to  $\geq$  {3400} psig is sufficient to ensure that there will be an adequate supply of breathable air to maintain MCR habitability for a period of 72 hours. The Frequency of 24 hours is based on the availability of pressure indication in the MCR.

1

SR 3.7.6.3

VES air delivery isolation valves are required to be verified as OPERABLE. The Frequency required is in accordance with the Inservice Testing Program.

SR 3.7.6.4

VES air header isolation valves are required to be verified open at 31 day intervals. This SR is designed to ensure that the pathways for supplying breathable air to the MCR are available should loss of VBS occur. These valves should be closed only during required testing or maintenance of downstream components, or to preclude complete depressurization of the system should the VES isolation valves in the air delivery line open inadvertently or begin to leak.

SR 3.7.6.5

Verification that the air quality of the air storage tanks meets the requirements of Appendix C, Table C-1 of ASHRAE Standard 62 is required every 92 days. If air has not been added to the air storage tanks since the previous verification, verification may be accomplished by confirmation of the acceptability of the previous surveillance results along with examination of the documented record of air makeup. The purpose of ASHRAE Standard 62 states: "This standard specifies minimum ventilation rates and indoor air quality that will be acceptable to human occupants and are intended to minimize the potential for adverse health

BASES

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ACTIONS (continued)

11

If the charger is operating in the current limit mode after 6 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 24 hours (Required Action A.2).

Required Action A.2 requires that the battery float current be verified as less than or equal to [5] amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it has now been fully recharged. If at the expiration of the initial 24 hour period the battery float current is not less than or equal to [5] amps this indicates there may be additional battery problems and the battery must be declared inoperable.

Required Action A.3 limits the restoration time for the inoperable battery charger to 7 days. This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used (e.g., balance of plant non-Class 1E battery charger). The 7 day Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

B.1, B.2, and B.3

Condition B represents two divisions with one or more battery chargers inoperable (e.g., the voltage limit of SR 3.8.1.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action B.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 2 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within 24 hours, the battery will be restored to its fully charged condition (Required Action B.2) from any discharge that might have occurred due to the charger inoperability.

BASES

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ACTIONS (continued)

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 24 hours, avoiding a premature shutdown with its own attendant risk.

If the charger is operating in the current limit mode after 2 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 24 hours (Required Action B.2).

Required Action B.2 requires that the battery float current be verified as less than or equal to [5] amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it has now been fully recharged. If at the expiration of the initial 24 hour period the battery float current is not less than or equal to [5] amps this indicates there may be additional battery problems and the battery must be declared inoperable.

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Required Action B.3 limits the restoration time for the inoperable battery charger to 7 days. This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used (e.g., balance of plant non-Class 1E battery charger). The 7 day Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

C.1

Condition C represents one division with one or more batteries inoperable. With one or more batteries inoperable, the DC bus is being supplied by the OPERABLE battery chargers. Any event that results in a loss of the AC bus supporting the battery chargers will also result in loss of DC to that train.

BASES

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ACTIONS (continued)

Because of the passive system design and the use of fail-safe components, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate most DBAs following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one DC subsystem against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

The installed spare battery bank and charger may be used to restore an inoperable Class 1E DC electrical power subsystem; however, all applicable Surveillances must be met by the spare equipment used, prior to declaring the subsystem OPERABLE.

D.1

Condition D represents two divisions with one or more batteries inoperable. With one or more batteries inoperable, the DC bus is being supplied by the OPERABLE battery charger. Any event that results in a loss of the AC bus supporting the battery charger will also result in loss of DC to that train. The 2 hour limit allows sufficient time to effect restoration of an inoperable battery given that the majority of the conditions that lead to battery inoperability (e.g., loss of battery charger, battery cell voltage less than {2.07} V, etc.) are identified in Specifications 3.8.1, 3.8.2, and 3.8.7 together with additional specific completion times.

| (1)

The installed spare battery bank and charger may be used to restore an inoperable Class 1E DC electrical power subsystem; however, all applicable Surveillances must be met by the spare equipment used, prior to declaring the subsystem OPERABLE.

E.1

If one of the Class 1E DC electrical power subsystems is inoperable, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate all design basis accidents, based on conservative analysis.

Because of the passive system design and the use of fail-safe components, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate most DBAs

BASES

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ACTIONS (continued)

G.1 and G.2

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to 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 and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.1.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the battery chargers which support ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state while supplying the continuous steady state loads of the associated DC subsystem. On float charge, battery cells will receive adequate current to optimally charge the battery. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. This voltage maintains the battery plates in a condition that supports maintaining the grid life (expected to be approximately 20 years). The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 8).

SR 3.8.1.2

This SR verifies the design capacity of the battery chargers. According to Regulatory Guide 1.32 (Ref. 9), the battery charger supply is recommended to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensure that these requirements can be satisfied.

1

This SR provides two options. One option requires that each battery charger be capable of supplying [400] amps at the minimum established float voltage for [248] hours. The ampere requirements are based on the output rating of the chargers. The voltage requirements are based on the

BASES

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SURVEILLANCE REQUIREMENTS (continued)

1

charger voltage level after a response to a loss of AC power. The time period is sufficient for the charger temperature to have stabilized and to have been maintained for at least [2] hours.

The other option requires that each battery charger be capable of recharging the battery after a service test coincident with supplying the largest coincident demands of the various continuous steady state loads (irrespective of the status of the plant during which these demands occur). This level of loading may not normally be available following the battery service test and will need to be supplemented with additional loads. The duration for this test may be longer than the charger sizing criteria since the battery recharge is affected by float voltage, temperature, and the exponential decay in charging current. The battery is recharged when the measured charging current is  $\leq$  [2] amps.

11

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 24 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

SR 3.8.1.3

A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the Class 1E DC electrical power system. The discharge rate and test length corresponds to the design duty cycle requirements as specified in Reference 4.

The Surveillance Frequency of 24 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 9) and Regulatory Guide 1.129 (Ref. 10), which state that the battery service test should be performed with intervals between tests not to exceed 24 months. This Surveillance may be performed during any plant condition with the spare battery and charger providing power to the bus.

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test.

The modified performance discharge test is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity,

## B 3.8 ELECTRICAL POWER SYSTEMS

### B 3.8.2 DC Sources – Shutdown

#### BASES

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**BACKGROUND** A description of the Class 1E DC power sources is provided in the Bases for LCO 3.8.1, “DC Sources – Operating.”

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**APPLICABLE SAFETY ANALYSES** The initial conditions of Design Basis Accident (DBA) and transient analyses in the Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume engineered safety features are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the emergency auxiliaries and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystem is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems’ OPERABILITY.

The OPERABILITY of the minimum Class 1E DC power sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate Class 1E DC power sources are provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES {1, 2, 3, and 4} have no specific analyses in MODES {5 and 6} because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal

BASES

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ACTIONS (continued)

- c. The potential for an event in conjunction with a single failure of a redundant component.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an AC instrument and control bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 6 hours. This could lead to a total of 6 hours, since initial failure of the LCO, to restore the DC distribution system. At this time, an AC train could again become inoperable, and DC distribution restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal “time zero” for beginning the allowed outage time “clock.” This will result in establishing the “time zero” at the time the LCO was initially not met, instead of the time Condition B was entered. The 12 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1

With two divisions of AC instrument and control buses inoperable, the remaining OPERABLE buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the required divisions of AC instrument and control buses must be restored to OPERABLE status within 2 hours by powering the bus from the associated [inverter via inverted DC, inverter using internal AC source, or Class 1E constant voltage transformer].

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Condition C represents two divisions of AC instrument and control vital buses without power; potentially both the DC source and the associated AC source are nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all noninterruptable power. It is, therefore, imperative that the operator’s attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining buses and restoring power to the affected buses.



BASES

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ACTIONS (continued)

D.1

1

With two divisions of DC electrical power distribution subsystems inoperable, the remaining DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, the [required] DC buses must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery or charger.

Condition D represents two subsystems without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining divisions and restoring power to the affected divisions.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

- a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;
- b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected divisions; and
- c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 3).

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Battery Parameters

BASES

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BACKGROUND

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LCO 3.8.7, Battery Parameters, delineates the limits on electrolyte temperature, level, float voltage and specific gravity for the DC power source batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.1, "DC Sources – Operating," and LCO 3.8.2, "DC Sources – Shutdown." In addition to the limitations of this Specification, the [licensee controlled program] also implements a program specified in Specification 5.5.11 for monitoring various battery parameters that is based on the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice For Maintenance, Testing, And Replacement Of Vented Lead-Acid Batteries For Stationary Applications" (Ref. 3).

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APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in Chapter 6 (Ref. 1), and Chapter 15 (Ref. 2), assume engineered safety features are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for safety related and vital control instrumentation loads including monitoring and main control room emergency lighting during all MODES of operation. It also provides power for safe shutdown when all the onsite and offsite AC power sources are lost.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least three of the four Divisions of DC sources OPERABLE during accident conditions, in the event of:

- a. An assumed loss of all offsite and onsite AC power sources; and
- b. A worst case single failure.

Battery parameters satisfy the Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

Battery parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Battery parameter limits are conservatively established, allowing continued DC electrical system function even with

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BASES

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LCO (continued)

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limits not met. Additional preventative maintenance, testing, and monitoring performed in accordance with the [licensee controlled program] is conducted as specified in Specification 5.5.11.

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APPLICABILITY

The battery parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery parameter limits are only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.1, and LCO 3.8.2.

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ACTIONS

A.1, A.2, and A.3

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With one or more cells in one or more batteries in one Division  $< [2.07]$  V, the battery cell is degraded. Within 2 hours verification of the required battery charger, OPERABILITY is made by monitoring the battery terminal voltage (SR 3.8.1.1) and of the overall battery state of charge by monitoring the battery float charge current (SR 3.8.7.1). This assures that there is still sufficient battery capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of one or more cells in one or more batteries  $< [2.07]$  V, and continued operation is permitted for a limited period up to 24 hours.

Since the Required Actions only specify "perform," a failure of SR 3.8.1.1 or SR 3.8.7.1 acceptance criteria does not result in this Required Action not met. However, if one of the SRs is failed the appropriate Condition(s), depending on the cause of the failures, is entered. If SR 3.8.7.1 is failed then there is not assurance that there is still sufficient battery capacity to perform the intended function and the battery must be declared inoperable immediately.

B.1 and B.2

14

One or more batteries in one Division with float  $> [25]$  amps indicates that a partial discharge of the battery capacity has occurred. This may be due to a temporary loss of a battery charger or possibly due to one or more battery cells in a low voltage condition reflecting some loss of capacity. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage. If the terminal voltage is found to be less than the minimum established float voltage there are two possibilities, the battery charger is inoperable or is operating in the current limit mode. Condition A addresses charger inoperability. If the

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BASES

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ACTIONS (continued)

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charger is operating in the current limit mode after 2 hours that is an indication that the battery has been substantially discharged and likely cannot perform its required design functions. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within ~~{12}~~24 hours (Required Action B.2). The battery must therefore be declared inoperable.

If the float voltage is found to be satisfactory but there are one or more battery cells with float voltage less than {2.07} V, the associated "OR" statement in Condition F is applicable and the battery must be declared inoperable immediately. If float voltage is satisfactory and there are no cells less than {2.07} V there is good assurance that, within ~~{12}~~24 hours, the battery will be restored to its fully charged condition (Required Action B.2) from any discharge that might have occurred due to a temporary loss of the battery charger.

A discharged battery with float voltage (the charger setpoint) across its terminals indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within ~~{12}~~24 hours, avoiding a premature shutdown with its own attendant risk.

If the condition is due to one or more cells in a low voltage condition but still greater than {2.07} V and float voltage is found to be satisfactory, this is not indication of a substantially discharged battery and ~~{12}~~24 hours is a reasonable time prior to declaring the battery inoperable.

Since Required Action B.1 only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in the Required Action not met. However, if SR 3.8.1.1 is failed, the appropriate Condition(s), depending on the cause of the failure, is entered.

C.1, C.2, and C.3

With one or more batteries in one Division with one or more cells electrolyte level above the top of the plates, but below the minimum established design limits, the battery still retains sufficient capacity to

BASES

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ACTIONS (continued)

F.1

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14

With one or more batteries with any battery parameter outside the allowances of the Required Actions for Condition A, B, C, D, or E, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding battery must be declared inoperable. Additionally, discovering one or more batteries in one Division with one or more battery cells float voltage less than [2.07] V and float current greater than [25] amps indicates that the battery capacity may not be sufficient to perform the intended functions. The battery must therefore be declared inoperable immediately.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.7.1

Verifying battery float current while on float charge is used to determine the state of charge of the battery. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a charged state. The float current requirements are based on the float current indicative of a charged battery. Use of float current to determine the state of charge of the battery is consistent with IEEE-450 (Ref. 3). The 7 day Frequency is consistent with IEEE-450 (Ref. 3).

11

This SR is modified by a Note that states the float current requirement is not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.1.1. When this float voltage is not maintained the Required Actions of LCO 3.8.1 ACTION A are being taken, which provide the necessary and appropriate verifications of the battery condition. Furthermore, the float current limit of [2] amps is established based on the nominal float voltage value and is not directly applicable when this voltage is not maintained.

SR 3.8.7.2 and SR 3.8.7.5

1

Optimal long term battery performance is obtained by maintaining a float voltage greater than or equal to the minimum established design limits provided by the battery manufacturer, which corresponds to [130.5132.0] V at the battery terminals, or [2.250] Vpc. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge, which could eventually render the battery inoperable. Float voltages in this range or less, but greater than [2.07] Vpc, are addressed in Specification 5.5.11. SRs 3.8.7.2 and 3.8.7.5 require verification that the cell float voltages are equal to or greater than the short term absolute

BASES

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SURVEILLANCE REQUIREMENTS (continued)

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minimum voltage of {2.07} V. The Frequency for cell voltage verification every 31 days for pilot cell and 92 days for each connected cell is consistent with IEEE-450 (Ref. 3).

SR 3.8.7.3

The limit specified for electrolyte level ensures that the plates suffer no physical damage and maintains adequate electron transfer capability. The Frequency is consistent with IEEE-450 (Ref. 3).

SR 3.8.7.4

1

This Surveillance verifies that the pilot cell temperature is greater than or equal to the minimum established design limit (i.e., {4060}°F). Pilot cell electrolyte temperature is maintained above this temperature to assure the battery can provided the required current and voltage to meet the design requirements. Temperatures lower than assumed in battery sizing calculations act to inhibit or reduce battery capacity. The Frequency is consistent with IEEE-450 (Ref. 3).

SR 3.8.7.6

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.7.6; however, only the modified performance discharge test may be used to satisfy the battery service test requirements of SR 3.8.1.3.

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test.

It may consist of just two rates; for instance the one minute rate for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a

BASES

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SURVEILLANCE REQUIREMENTS (continued)

one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test must remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 3) and IEEE-485 (Ref. 4). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. Furthermore, the battery is sized to meet the assumed duty cycle loads when the battery design capacity reaches this {80}% limit.

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The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity  $\geq$  100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450 (Ref. 3), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is  $\geq$  {10%} below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 3).

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown

## B 3.9 REFUELING OPERATIONS

### B 3.9.5 Containment Penetrations

#### BASES

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#### BACKGROUND

During movement of irradiated fuel assemblies within containment, potential releases of fission product radioactivity within containment are monitored and filtered or are restricted from escaping to the environment when the LCO requirements are met. Monitoring of potential releases of radiation is performed in accordance with Administrative Controls Section 5.5.2, "Radioactive Effluent Control Program." In MODES 1, 2, 3, and 4, containment OPERABILITY is addressed in LCO 3.6.1, "Containment." In MODES 5 and 6, closure capability of containment penetrations is addressed in LCO 3.6.8, "Containment Penetrations." Since there is no potential for containment pressurization due to a fuel handling accident, the Appendix J leakage criteria and tests are not required in MODES 5 and 6.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within the requirements of 10 CFR 50.34. For a fuel handling accident, the AP1000 dose analysis does not rely on containment closure to meet the offsite radiation exposure limits. This LCO is provided as an additional level of defense against the possibility of a fission product release from a fuel handling accident.

The containment equipment hatches, which are part of the containment pressure boundary, provide a means for moving large equipment and components into and out of containment. During movement of irradiated fuel assemblies within containment, an equipment hatch is considered closed if the hatch cover is held in place by at least {four} bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

7

If the equipment hatch is open, an alternative barrier between the containment atmosphere and the outside atmosphere shall be in place. Each containment equipment hatch opens into a staging area in the auxiliary building. These staging areas contain doors that open to the radiologically controlled areas of the annex building. The annex building contains a door that opens to the outside atmosphere. The alternate barrier may consist of the staging area in the auxiliary building, or may consist of the staging areas in the auxiliary building and the radiologically controlled areas in the annex building provided the doors from the annex building to the outside atmosphere are closed. The alternate barrier may



BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.5.3

7

This SR verifies the ability of the VFS to maintain a negative pressure ( $\leq \{-0.125\}$  inches water gauge relative to outside atmospheric pressure) in the containment and the portions of the auxiliary and/or annex building that comprise the envelope defined as the alternate barrier. This surveillance is performed with the VFS in containment operating. Doors in the alternate barrier which are normally closed may be opened for ingress and egress. The portion of the VAS which services the area enclosed by the alternate barrier is aligned to the VFS exhaust subsystem, and the VAS auxiliary/annex building supply fans and VFS containment purge supply fans not operating. The Frequency of 24 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 3).

SR 3.9.5.4

The VFS should be checked periodically to ensure that it functions properly. As the operating conditions on this system are not severe, testing each train within 31 days prior to fuel movement provides an adequate check on this system. Operation of the heater dries out any moisture accumulated in the charcoal from humidity in the ambient air.

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REFERENCES

1. Section 15.7.4, "Fuel Handling Accident."
  2. NUREG-0800, Section 15.0.1, Rev. 0.
  3. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.6 Containment Air Filtration System (VFS)

#### BASES

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#### BACKGROUND

The radiologically controlled area ventilation system (VAS) serves the fuel handling area of the auxiliary building, and the radiologically controlled portions of the auxiliary and annex buildings, except for the health physics and hot machine shop areas which are provided with a separate ventilation system (VHS). If high airborne radioactivity is detected in the exhaust air from the fuel handling area, the auxiliary building, or the annex buildings, the VAS supply and exhaust duct isolation dampers automatically close to isolate the affected area from the outside environment and the containment air filtration exhaust subsystem starts. The VFS exhaust subsystem prevents exfiltration of unfiltered airborne radioactivity by maintaining the isolated zone at  $\leq \{-0.125\}$  inches water gauge pressure relative to the outside atmosphere. Monitoring of potential releases of radiation is performed in accordance with Administrative Controls Section 5.5.2, "Radioactive Effluent Control Program."

7

For a fuel handling accident, the AP1000 dose analysis does not rely on the OPERABILITY of the VAS or VFS exhaust subsystem to meet the offsite radiation exposure limits. This LCO is provided as an additional level of defense-in-depth against the possibility of a fission product release from a fuel handling accident in the fuel building. The plant vent radiation detectors monitor effluents discharged from the plant vent to the environment.

Each VFS exhaust subsystem includes one 100 percent capacity exhaust air filtration unit, and the associated exhaust fan, heater and ductwork.

The filtration units are connected to a ducted system with isolation dampers to provide HEPA filtration and charcoal adsorption of exhaust air from the containment, fuel handling area, radiologically controlled areas of the auxiliary and annex buildings. A gaseous radiation monitor is located downstream of the exhaust air filtration units to provide an alarm if abnormal gaseous releases are detected. The plant vent exhaust flow is monitored for gaseous, particulate and iodine releases to the environment. During conditions of abnormal airborne radioactivity in the fuel handling area, auxiliary and/or annex buildings, the VFS exhaust subsystem provides filtered exhaust to minimize unfiltered offsite releases.

## BASES

## ACTIONS (continued)

LCO 3.0.8 is applicable while in MODE 5 or 6. Since irradiated fuel assembly movement can occur in MODE 5 or 6, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, the fuel movement is independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

A.1

When the required VFS exhaust subsystem is inoperable during movement of irradiated fuel assemblies in the fuel building, action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE  
REQUIREMENTSSR 3.9.6.1

Each VFS exhaust subsystem should be checked 31 days prior to fuel movement in the fuel handling area to ensure that it functions properly. As the operating conditions on this subsystem are not severe, testing each subsystem within one month prior to fuel movement provides an adequate check on this system. Operation of the heater dries out any moisture accumulated in the charcoal from humidity in the ambient air.

SR 3.9.6.2

This SR verifies that the VAS fuel handling area subsystem aligns to the VFS and that the VFS exhaust subsystem starts and operates on an actual or simulated actuation signal. During the post-accident mode of operation, the VAS fuel handling area subsystem aligns to the VFS filtered exhaust subsystem. The 24 month Frequency is consistent with Reference 4.

SR 3.9.6.3

This SR verifies the integrity of the fuel handling area of the auxiliary building enclosure. The ability of the VAS and VFS to maintain negative pressure ( $\leq \{-0.125\}$  inches water gauge relative to outside atmospheric pressure) in the fuel handling area of the auxiliary building is periodically tested to verify proper function of the VAS and VFS exhaust subsystem. During this surveillance, the VAS fuel handling area subsystem is aligned

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BASES

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SURVEILLANCE REQUIREMENTS (continued)

7

to the operating VFS exhaust subsystem. The fan for the VAS fuel handling area subsystem is off. In this configuration, the VFS exhaust subsystem is designed to maintain a negative pressure in the fuel handling area of the auxiliary building ( $\leq \{-0.125\}$  inches water gauge relative to outside atmospheric pressure), to prevent unfiltered and unmonitored leakage. Doors may be opened for short periods of time to allow ingress and egress. During this surveillance, the VAS may be servicing the remaining portions of the auxiliary and annex buildings. The Frequency of 24 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 5).

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REFERENCES

1. Section 9.4.3, "Radiologically Controlled Area Ventilation System."
  2. Section 9.4.7, "Containment Air Filtration System."
  3. Section 15.7.4, "Fuel Handling Accident."
  4. Regulatory Guide 1.140 (Rev. 2).
  5. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.
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