REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**154-8064**
SRP Section:	**16 - Technical Specification**
Application Section:	**16 - Technical Specification**
Date of RAI Issue:	**08/17/2015**

Question No. 16-44

Paragraph (a)(11) of 10 CFR 52.47 and paragraph (a)(30) of 10 CFR 52.79 state that a design certification (DC) applicant and a combined license (COL) applicant, respectively, are to propose TS prepared in accordance with 10 CFR 50.36 and 50.36a.
DCD Tier 2 Section 16.1.2.4 states "Single brackets ([ ]) are used to identify the preliminary design information or plant-specific information. Double brackets ([[ ]]) indicate the conceptual design information for those portions of the plant for which the application does not seek certification." SRP Section 16.0 explains that COL action items, also referred to as site-specific information, are indicated in the generic technical specifications (TS) and Bases, DCD Tier 2 Chapter 16, usually by use of square brackets. Section 182a of the Atomic Energy Act requires TS to be included with any operating license for a utilization facility issued by the NRC. Consequently, the plant-specific TS issued with a COL must be complete and useable for facility operation. Therefore, a COL applicant must resolve all COL action items in the generic TS and Bases in order to complete the plant-specific TS for issuance with the COL in accordance with 10 CFR 52.97.Since it is possible for "conceptual design information" to not be finalized until after COL issuance, generic TS and Bases cannot contain placeholders for such information. The applicant is requested to revise DCD Tier 2 Section 16.2.4 to omit discussion of the possible use of double bracketed conceptual design information, and delete any placeholders for such information from the generic TS and Bases, or replace it with placeholders for site-specific information in square brackets, which can be finalized by a COL applicant before COL issuance. (Staff observed that double brackets are only used in generic TS 3.7.9, Ultimate Heat Sink.)

In addition, the applicant is requested to provide a list of the Chapter 16 COL Action Items, providing a concise description of each. Staff suggests enumerating each action item using a prefix consisting of either (a) the numerical label designation of the affected generic TS section or subsection, that contains the bracketed TS information (e.g., COL Action Item 3.8.1-1, 3.3.1-3, 2.0-1, 1.1-2, 5.5.4-2); or (b) the alpha numerical designation of the affected generic TS Bases section or subsection (e.g., B 3.8.1-1, B 3.3.1-3, B 2.1.2-1, B 3.0-1, B

3.6.3-2, etc.), that contains the bracketed TS Bases information. To the prefix append a hyphen and a sequential number of the item in that section or subsection, as appropriate.

As necessary, provide guidance to clarify expectations for properly completing or resolving each COL action item needing such guidance. This guidance has been presented by previous design certification applications as bracketed reviewer's notes in the generic TS Bases or in a table listing the action items located in the introductory part of DCD Tier 2 Chapter 16.

Response

KHNP will replace double brackets to single brackets used in generic TS 3.7.9, Ultimate Heat Sink as shown in attached markup. The definition of the double brackets will be removed from section 16.1.2.4 Combined License Information.

KHNP will provide a list of the COL Action Items with a concise description of each item in Section 16.1.2.4. as shown in the attached markup.

Feedback on response to Question 16-44

Staff believes there are places in the generic TS and Bases that have bracketed information other than Subsections 3.7.11, 3.8.1, and 3.8.3 (the only ones listed in the attachment to the response). Also, Subsection 3.7.9 should also be listed in Table 16-1.
Staff asks that KHNP to do a bracket search of Chapter 16 to verify it has not missed any other occurrences of COL information. Also request that KHNP decide how it will enumerate its COL information; at the very least, the DCD Chapter 1 should list one COL Action Item for Chapter 16; e.g., 16-1. The title for Table 16-1 should say “List of COL Action Items”

Response – (Rev.2)

KHNP verified all the bracketed information in DCD Tier 2 Section 16 based on Rev.1. List of COL Action Items of Table 16-1 in the DCD Rev.1 will be updated to add COL Action Items with a concise description of each item in other Subsections as shown in the attached markup. Table 1.8-2 in the DCD Chapter 1 will be also revised to add COL Action Items in the generic TS and Bases as indicated in the attached markup.

The Steam Generator repair method of TS 5.5.9 and TS 5.6.7, and all Reviewer’s Notes will be bracketed as COL Action Items as indicated in the attached markup.

Impact on DCD

Same as changes described in Impact on Technical Specifications section.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

Technical Specification 16.1.2.4 and 3.7.9 Ultimate Heat Sink LCO and Bases sections will be revised as shown in attached markup.

DCD Table 16-1 of Chapter 16, Table 1.8-2 of Chapter 1, Subsection 16.1.1, TS 3.1.8, TS 3.5.4, TS 3.8.1, TS 5.1, TS 5.1.2, TS 5.2.2, TS 5.3, TS 5.5.3, TS 5.5.9, TS 5.5.11, TS 5.5.17, TS 5.5.19, TS 5.6.4, TS 5.6.5, TS 5.6.7, Bases 3.0.9, Bases 3.7.11, Bases 3.7.12, Bases 3.8.1, and Bases 3.9.3 will be revised as shown in the attached markup.

Impact on Technical/Topical/Environmental Report

There is no impact on any Technical, Topical, or Environment Reports.

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Table 16-1 (1 of 2)

Replace with 'A' in the next pages

List of COL Action Item

TS Section	Description	Resolution
3.7.11	Control Room Habitability Area option for design features to protect occupant exposures to toxic gases	The specific toxic gas concentrations in the air intakes will vary depending on site. If the applicant determines that the maximum concentrations for the air intakes for a given toxic gases do not exceed the toxicity limits from Regulatory Guide 1.78 prior to 2 minutes, toxic gas detector is not required and the bracketed phrases are deleted.
3.8.1	SR 3.8.1.4, Day Tank Capacity	The specific value will vary depending on engine manufacturer's specific design recommendations.
	SR 3.8.1.8, Offsite Power Transfer Test	Plant operation MODES which allow the Surveillance depend on the plant operation and surveillance policy.
	SR 3.8.1.9, Emergency Diesel Generator (EDG) Single Largest Load Rejection Test	1) Plant operation MODES which allow the Surveillance depend on the plant operation and surveillance policy. 2) EDG operation power factor depends on plant specific EDG technical specification.
	SR 3.8.1.10, EDG Full-Load Rejection Test	1) Plant operation MODES which allow the Surveillance depend on the plant operation and surveillance policy. 2) EDG operation power factor depends on plant specific EDG technical specification.
	SR 3.8.1.12, EDG Engineered Safety Features (ESF) Actuation Test	Plant operation MODES which allow the Surveillance depend on the plant operation and surveillance policy.
	SR 3.8.1.13, EDG Bypassed Trip Signal Test	Plant operation MODES which allow the Surveillance depend on the plant operation and surveillance policy.
	SR 3.8.1.14, EDG Endurance and Load Test	1) Plant operation MODES which allow the Surveillance depend on the plant operation and surveillance policy. 2) EDG operation power factor depends on plant specific EDG technical specification.

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Replace with 'A' in the next pages

Table 16-1 (2 of 2)

TS Section	Description	Resolution
3.7.11	Control Room Habitability Area option for design features to protect occupant exposures to toxic gases	The specific toxic gas concentrations in the air intakes will vary depending on site. If the applicant determines that the maximum concentrations for the air intakes for a given toxic gases do not exceed the toxicity limits from Regulatory Guide 1.78 prior to 2 minutes, toxic gas detector is not required and the bracketed phrases are deleted.
3.8.3	Actions E, Surveillance Requirement 3.8.3.4 Starting Air Receiver Pressure	The air pressure of the starting air receiver will vary depending on engine manufacturer's specific design recommendations.

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Insert 'C' in the next page before COL 16-3.6(1)

List of COL Action Items

Item No.	TS Section	Description	Resolution
COL 16-3.6(1)	3.6.7	Containment Penetrations – Shutdown Operations	The COL applicant is to provide the minimum number of bolt, completion time, and surveillance frequency for shutdown operations. The value will be determined based on plant specific Shutdown Evaluation Report to satisfy the 10 CFR 50.34 dose limits.
COL 16-3.7(1)	3.7.9	Ultimate Heat Sink	The COL applicant is to provide the completion time, and surveillance frequency for ultimate heat sink. Ultimate heat sink design value varies depending on site characteristics.
COL 16-3.7(2)	3.7.11	Control Room Habitability Area option for design features to protect occupant exposures to toxic gases	The COL applicant is to provide the details of specific toxic chemicals of mobile and stationary sources and evaluate the MCR habitability based on the recommendation in RG 1.78. The specific toxic gas concentrations in the air intakes will vary depending on site. If the applicant determines that the maximum concentrations in the MCR for a given toxic gases do not exceed the toxicity limits from RG 1.78, toxic gas detector is not required and the bracketed phrases are deleted.
COL 16-3.8(1)	3.8.1	SR 3.8.1.4, Day Tank Capacity	The COL applicant is to provide the specific value in accordance with EDG manufacture's specific design characteristics.
COL 16-3.8(2)	3.8.1	SR 3.8.1.8, Offsite Power Transfer Test SR 3.8.1.9, EDG Single Largest Load Rejection Test SR 3.8.1.10, EDG Full-Load Rejection Test SR 3.8.1.12, EDG ESF Actuation Test SR 3.8.1.13, EDG Bypassed Trip Signal Test	The COL applicant is to determine plant operation MODES which allow the Surveillance depend on the plant operation and surveillance policy. The MODES restrictions may be deleted if the COL applicant demonstrates that the plant safety is maintained or enhanced when the surveillance is performed in restricted MODES
COL 16-3.8(3)	3.8.1	SR 3.8.1.9, EDG Single Largest Load Rejection Test SR 3.8.1.10, EDG Full-Load Rejection Test SR 3.8.1.14, EDG Endurance and Load Test	The COL applicant is to determine EDG power factor as applicable to the tests. EDG operation power factor depends on plant specific EDG Class 1E loads and offsite power condition.
COL 16-3.8(4)	3.8.3	Actions E, SR 3.8.3.4 Starting Air Receiver Pressure	The COL applicant is to provide the specific value in accordance with EDG manufacturer's specific design recommendation.

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COL 16-3.9(1)	3.9.3	Containment Penetration	The COL applicant is to provide the minimum number of bolt for equipment hatch. The value will be determined based on plant specific Shutdown Evaluation Report to satisfy the 10 CFR 50.34 dose limits.
COL 16-4.1(1)	4.1	Site Location	Information on site location is to be provided by the COL applicant
COL 16-5.3(1)	5.3	Unit Staff Qualification	The requirement for unit staff qualification shall be determined by COL applicant based on latest NRC RG 1.8 and ANSI standard acceptable to NRC staff.
COL 16-5.4(1)	5.4	Procedure	The COL applicant will determine the modification of core protection calculator (CPC) addressable constants based on plant specific data.
~~COL 16-5.5(1)~~	~~5.5.3~~	~~Post-Accident Sampling~~	~~Information on licensee is to be provided by the COL applicant~~
~~COL 16-5.5(2)~~	~~5.5.3~~	~~Post-Accident Sampling~~	~~Information on plant is to be provided by the COL applicant~~
~~COL 16-5.5(3)~~	~~5.5.11~~	~~Ventilation Filter Testing Program~~	~~Information on plant specific allowable penetration equation is to be provided by the COL applicant~~
COL 16-5.5(~~4~~) 1	5.5.12	Explosive Gas and Storage Tank Radioactivity Monitoring Program	The methodology for gaseous radioactivity quantities and the liquid radwaste quantities is to be provided by the COL applicant
COL 16-5.5(~~5~~) 2	5.5.19	Setpoint Control Program	The FSAR reference on setpoint control document is to be specified by the COL applicant
COL 16-5.6(1)	5.6.1 5.6.2	Annual Radiological Environmental Operating Report Radiological Effluent Release Report	A single submittal of reporting on multiple unit stations is to be determined in COL stage.
COL 16-5.6(2)	5.6.1	Annual Radiological Environmental Operating Report	The COL applicant will determine the format of the table. Either format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979, or use another format that is acceptable to NRC
COL 16-5.6(3)	5.6.7	Steam Generator Tube Inspection Report	The COL applicant is to provide repair method utilized and number of tubes repaired by each repair method.

Insert 'D' in the next page after COL 16-4.1(1)

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C

COL 16-3(1)	B3.0.9 B3.7.11 B3.7.12 B3.8.1 B3.9.3	LCO Applicability Control Room HVAC System Auxiliary Building Controlled Area Emergency Exhaust System AC Sources - Operating Containment Penetrations	Applicability of Reviewer's Note is to be determined by the COL applicant.

D

COL 16-5(1)	5.1 5.2.2 5.3 5.5.3 5.5.9 5.5.11 5.5.17 5.5.19 5.6.4	Responsibility Unit Staff Unit Staff Qualifications Post-Accident Sampling Steam Generator(SG) Program Ventilation Filter Testing Program Battery Monitoring and Maintenance Program Setpoint Control Program Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT(PTLR)	Applicability of Reviewer's Note is to be determined by the COL applicant.

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Table 1.8-2 (32 of 38)

Item No.	Description
COL 14.2(11)	The COL applicant is to provide a schedule for the development of plant procedures, as well as a description of how, and to what extent, the plant operating, emergency, and surveillance procedures are use-tested during the initial test program.
COL 14.2(12)	The COL applicant that references the APR1400 design certification is to identify the specific operator training to be conducted as part of the low-power testing program related to the resolution of TMI Action Plan Item I.G.1, as described in (1) NUREG-0660 – NRC Action Plans Developed as a Result of the TMI-2 Accident, Revision 1, August 1980 and (2) NUREG-0737 – Clarification of TMI Action Plan Requirements.
COL 14.2(13)	The COL applicant is to develop a sequence and schedule for the development of the plant operating and emergency procedures should allow sufficient time for trial use of these procedures during the Initial Test Program. The sequence and schedule for plant startup is to be developed by the COL applicant to allow sufficient time to systematically perform the required testing in each phase.
COL 14.2(14)	The COL applicant is to perform the appropriate interface testing of the gaseous PERMSS monitors with ERDS.
COL 14.2(15)	The COL applicant is to prepare the preoperational test of cooling tower and associated auxiliaries, and raw water and service water cooling systems.
COL 14.2(16)	The COL applicant is to develop the test program of personnel monitors, radiation survey instruments, and laboratory equipment used to analyze or measure radiation levels and radioactivity concentrations.
COL 14.2(17)	The COL applicant is to prepare the site-specific preoperational and startup test specification and test procedure and/or guideline for plant and offsite communication system.
COL 14.2(18)	The COL applicant is to prepare the pre-operational test of ultimate heat sink pump house.
COL 14.2(19)	The COL applicant is to prepare the testing and verification of ultimate heat sink cooling chains.
COL 14.3(1)	The COL applicant is to provide the ITAAC for the site-specific portion of the plant systems specified in Subsection 14.3.3.
COL 14.3(2)	The COL applicant is to provide a design ITAAC closure schedule for implementing the V&V design ITAAC as addressed in Subsection 14.3.2.9.
COL 14.3(3)	The COL applicant is to provide the proposed ITAAC for the facility's emergency planning not addressed in the DCD in accordance with RG 1.206.
COL 14.3(4)	The COL applicant is to provide the proposed ITAAC for the site specific facility's physical security hardware not addressed in the DCD in accordance with RG 1.206.
COL 15.0(1)	The COL applicant is to perform the radiological consequence analysis using site-specific χ/Q values, unless the χ/Q values used in the DCD envelop the site-specific short-term or long-term χ/Q values of the DCD, and to show that the resultant doses are within the guideline values of 10 CFR 50.34 for EAB and LPZ and that of 10 CFR Part 50, Appendix A, GDC 19 for the MCR and TSC.
COL 16.1(1)	The choice of units is a COL information to be resolved by COL applicant

Insert "B" in the next pages after COL 16.1(1)

B RAI 154-8064_Q.16-44_Rev.1 RAI 154-8064_Q.16-44_Rev.2

Item No.	Description
COL 16-3.6(1)	The COL applicant is to provide the minimum number of bolt, completion time, and surveillance frequency for shutdown operations. The value will be determined based on plant specific Shutdown Evaluation Report to satisfy the 10 CFR 50.34 dose limits.
COL 16-3.7(1)	The COL applicant is to provide the completion time, and surveillance frequency for ultimate heat sink. Ultimate heat sink design value varies depending on site characteristics.
COL 16-3.7(2)	The COL applicant is to provide the details of specific toxic chemicals of mobile and stationary sources and evaluate the MCR habitability based on the recommendation in RG 1.78. The specific toxic gas concentrations in the air intakes will vary depending on site. If the applicant determines that the maximum concentrations in the MCR for a given toxic gases do not exceed the toxicity limits from RG 1.78, toxic gas detector is not required and the bracketed phrases are deleted.
COL 16-3.8(1)	The COL applicant is to provide the specific value in accordance with EDG manufacture's specific design characteristics.
COL 16-3.8(2)	The COL applicant is to determine plant operation MODES which allow the Surveillance depend on the plant operation and surveillance policy. The MODES restrictions may be deleted if the COL applicant demonstrates that the plant safety is maintained or enhanced when the surveillance is performed in restricted MODES
COL 16-3.8(3)	The COL applicant is to determine EDG power factor as applicable to the tests. EDG operation power factor depends on plant specific EDG Class 1E loads and offsite power condition.
COL 16-3.8(4)	The COL applicant is to provide the specific value in accordance with EDG manufacturer's specific design recommendation.
COL 16-3.9(1)	The COL applicant is to provide the minimum number of bolt for equipment hatch. The value will be determined based on plant specific Shutdown Evaluation Report to satisfy the 10 CFR 50.34 dose limits.
COL 16-4.1(1)	Information on site location is to be provided by the COL applicant
COL 16-5.3(1)	The requirement for unit staff qualification shall be determined by COL applicant based on latest NRC RG 1.8 and ANSI standard acceptable to NRC staff.
COL 16-5.4(1)	The COL applicant will determine the modification of core protection calculator (CPC) addressable constants based on plant specific data.
~~COL 16-5.5(1)~~	~~Information on licensee is to be provided by the COL applicant~~
~~COL 16-5.5(2)~~	~~Information on plant is to be provided by the COL applicant~~
~~COL 16-5.5(3)~~	~~Information on plant specific allowable penetration equation is to be provided by the COL applicant~~
COL 16-5.5(~~4~~) 1	The methodology for gaseous radioactivity quantities and the liquid radwaste quantities is to be provided by the COL applicant
COL 16-5.5(~~5~~) 2	The FSAR reference on setpoint control document is to be specified by the COL applicant
COL 16-5.6(1)	A single submittal of reporting on multiple unit stations is to be determined in COL stage.
COL 16-5.6(2)	The COL applicant will determine the format of the table. Either format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979, or use another format that is acceptable to NRC
COL 16-5.6(3)	The COL applicant is to provide repair method utilized and number of tubes repaired by each repair method.
COL 16-3(1)	Applicability of Reviewer's Note is to be determined by the COL applicant.
COL 16-5(1)	Applicability of Reviewer's Note is to be determined by the COL applicant.

CHAPTER 16 – TECHNICAL SPECIFICATIONS

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16.1 Introduction to Technical Specifications

16.1.1 Technical Specification Content

The content of the APR1400 Technical Specifications (TS) meets the requirements of 10 CFR 50.36. The APR1400 Technical Specifications were developed using the most appropriate guidance, NUREG-1432 Rev. 4.0 (Ref. 1).

The difference between NUREG-1432 and the APR1400 Technical Specification only exists as necessary to reflect advanced design features and operational features. The units specified in the APR1400 Technical Specifications are the International system of units (SI units) and the English units. The SI units have been used as the primary unit and the English units have been used in parentheses. The choice of units is a Combined License (COL) information item to be resolved by a COL applicant; however the TS and Bases do not enclose the parameter value pairs in square brackets. This is an exception to the use of brackets to denote COL information in the TS and Bases.

(COL 16.1(1))

16.1.1.1 Completion Times and Surveillance Frequencies

The Completion Times and Surveillance Frequencies specified in NUREG-1432 have generally applied to the associated Actions and Surveillance Requirements of the APR1400 Technical Specifications. For unique systems and features of the APR1400 design, similar Completion Times and Surveillance Frequencies have been adopted as appropriate.

16.1.1.2 Plant Design Difference

There are some design differences between the APR1400 Technical Specifications and current design in NUREG-1432. Major design differences include the four train emergency core cooling system design, the adoption of pilot operated safety relief valves (POSRVs), the change of ventilation systems, and auxiliary feedwater system configuration.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Charging Flow

LCO 3.1.8 Charging flow shall be maintained below 567.8 L/min (150 gpm) by closing charging flow restriction orifice bypass valves (CV-576, CV-577) and removing the power to the charging flow restriction orifice bypass valves.

APPLICABILITY: MODE 5 with reactor vessel level ≤ 36.3 m (119 ft 1 in) ~~[~~(hot leg level indication ≤ 100%~~]~~).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One of the required charging flow restriction orifice bypass valves not closed. OR One of the required charging flow restriction orifice bypass valves with power not removed.	A.1 Close CV-575 manually	Immediately
B. Both of required charging flow restriction orifice bypass valves not closed. OR Both of required charging flow restriction orifice bypass valves with power not removed.	--------------------NOTE-------------------- Auxiliary charging pump operation is allowed. --- B.1 Turn off all charging pump.	Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE			FREQUENCY
SR	3.5.4.1	Verify IRWST water temperature is ≥ 10°C (50°F) and ≤ 49°C (120°F).	24 hours
SR	3.5.4.2	Verify IRWST water volume is ≥ ~~[~~2,373.5 m^3 (627,000 gal)~~]~~ and ≤ ~~[~~2,540.6 m^3 (671,162 gal)~~]~~ (i.e., ≥ 74.43% and ≤ 79.67%).	7 days
SR	3.5.4.3	Verify IRWST boron concentration is ≥ 4,000 ppm and ≤ 4,400 ppm.	7 days

Brackets are deleted

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	------------------------------ NOTES ------------------------------ [1. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.] 2. If performed with EDG synchronized with offsite power, it shall be performed at a power factor ≤ ~~0.9~~ [0.9]. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. --- Verify each EDG does not trip, and voltage is maintained ≤ 4,576 V during and following a load rejection of ≥ 90% rating and ≤ 100% rating.	 18 months

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.12	------------------------------NOTES------------------------------ 1. ~~[~~All EDG starts may be preceded by an engine prelube period.~~]~~ 2. [This Surveillance shall not be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.] --	Brackets are deleted
	Verify on an actual or simulated Engineered Safety Features (ESF) actuation signal each EDG auto-starts from standby condition and: a. In ≤ 17 seconds after auto-start and during tests, achieves voltage ≥ 3,744 V and frequency ≥ 58.8 Hz, b. Achieves steady state voltage ≥ 3,744 V and ≤ 4,576 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz, c. Operates for ≥ 5 minutes, d. Permanently connected loads remain energized from the offsite power system, and e. Emergency loads are energized or auto-connected through the automatic load sequencer from the offsite power system.	18 months

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.19	-------------------------------NOTES------------------------------ 1. ~~[~~All EDG starts may be preceded by an engine prelube period.~~]~~ 2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---	Brackets are deleted
	Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses, b. Load shedding from emergency buses, c. EDG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 19 seconds, 2. energizes auto-connected emergency loads through load sequencer, 3. achieves steady state voltage ≥ 3,744 V and ≤ 4,576 V, 4. achieves steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.	18 months

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5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

Brackets are added

---REVIEWER'S NOTES---

[1. Titles for members of the unit staff shall be specified by use of an overall statement referencing an ANSI Standard acceptable to the NRC staff from which the titles were obtained, or an alternative title may be designated for this position. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special titles because of unique organizational structures.

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

5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence. The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety.

Brackets are deleted

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. 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.

5.2 Organization

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5.2.2 Unit Staff

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.

Brackets are added

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

[Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.]

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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 e 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. The operations manager or assistant operations manager shall hold an Senior Reactor Operator (SRO) license.

e. 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.

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5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

Brackets are added

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

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of [Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. [The staff not covered by NRC RG 1.8 shall meet or exceed the minimum qualifications of Regulations, NRC RGs, 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 Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

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5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Containment Spray System, Safety Injection System, Chemical and Volume Control System, Gaseous Waste Management System and Containment Hydrogen Control System. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements and

b. Integrated leak rate test requirements for each system at least once per 18 months.

The provisions of SR 3.0.2 are applicable.

5.5.3 Post-Accident Sampling

Brackets are added

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

[This program may be eliminated based on the implementation of Topical Report CE NPSD-1157, Rev. 1, "Technical Justification for the Elimination of the Post-Accident Sampling System from the Plant Design and Licensing Basis for CEOG Utilities," and the associated NRC Safety Evaluation dated May 16, 2000, and implementation of the following commitments:

1. ~~[~~Licensee~~]~~ has developed contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. The contingency plans will be contained in emergency plan implementing procedures and implemented with the implementation of the License amendment. Establishment of contingency plans is considered a regulatory commitment.

Brackets are deleted

Brackets are deleted

2. The capability for classifying fuel damage events at the Alert level threshold has been established for ~~[~~PLANT~~]~~ at radioactivity levels of 300 mCi/cc dose equivalent iodine. This capability may utilize the normal sampling system and/or correlations of sampling or letdown line dose rates to coolant concentrations. This capability will be described in emergency plan implementing procedures and implemented with the implementation of the License amendment. The capability for classifying fuel damage events is considered a regulatory commitment.

3. ~~[~~Licensee~~]~~ has established the capability to monitor radioactive iodines that have been released to offsite environs. This capability is described in our emergency plan implementing procedures. The capability to monitor radioactive iodines is considered a regulatory commitment.]

Brackets are deleted

5.5 Programs and Manuals

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5.5.9 Steam Generator (SG) Program (continued)

2. Accident induced LEAKAGE performance criterion: The primary-to-secondary accident-induced LEAKAGE rate for any design basis accident, other than a SG tube rupture, shall not exceed the LEAKAGE rate assumed in the accident analysis in terms of total LEAKAGE rate for all SGs and leakage rate for an individual SG. LEAKAGE is not to exceed 1.14 L/min (0.3 gpm) per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.12, "RCS Operational LEAKAGE."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

Brackets are added

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

[Alternate tube repair criteria currently permitted by plant technical specifications are listed here. The description of these alternate tube repair criteria should be equivalent to the descriptions in current technical specifications and should also include any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube repair criteria.]

----

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

5.5 Programs and Manuals

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5.5.9 Steam Generator (SG) Program (continued)

Brackets are added

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

[Plants are to include the appropriate Frequency (e.g., select the appropriate Item 2.) for their SG design. The first Item 2 is applicable to SGs with Alloy 600 mill annealed tubing. The second Item 2 is applicable to SGs with Alloy 600 thermally treated tubing. The third Item 2 is applicable to SGs with Alloy 690 thermally treated tubing.]

----

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

   a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

   b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

5.5 Programs and Manuals

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5.5.9 Steam Generator (SG) Program (continued)

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods."

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic nondestructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-to-secondary LEAKAGE.

f. [Provisions for SG tube repair methods. Steam generator tube rapair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

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

Tube repair methods currently permitted by plant technical specifications are to be listed here. The description of these tube repair methods should be equivalent to the descriptions in current technical specifications. If there are no approved tube repair methods, this section should not be use.

----

1. ....]

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables,

b. Identification of the procedures used to measure the values of the critical

5.5 Programs and Manuals

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5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

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

[The use of any standard other than ASTM D3803-1989 to test the charcoal sample may result in an overestimation of the capability of the charcoal to adsorb radioiodine. As a result, the ability of the charcoal filters to perform in a manner consistent with the licensing basis for the facility is indeterminate.

ASTM D 3803-1989 is a more stringent testing standard because it does not differentiate between used and new charcoal, it has a longer equilibration period performed at a temperature of 30°C (86°F) and a relative humidity (RH) of 95% (or 70% RH with humidity control), and it has more stringent tolerances that improve repeatability of the test.

Brackets are added

Brackets are deleted

Allowable Penetration = ~~[~~(100% - Methyl Iodide Efficiency * for Charcoal Credited in Licensee's Accident Analysis) / Safety Factor~~]~~

When ASTM D3803-1989 is used with 30°C (86°F) and 95% RH (or 70% RH with humidity control) is used, the staff will accept the following:

Safety factor 2 for systems with or without humidity control.

Humidity control can be provided by heaters or an NRC-approved analysis that demonstrates that the air entering the charcoal will be maintained less than or equal to 70 percent RH under worst case design basis conditions.

If the system has a face velocity greater than 110 percent of 0.203 m/s (40 ft/min), the face velocity should be specified.

*This value should be the efficiency that was incorporated in the licensee's accident analysis which was reviewed and approved by the staff in a safety evaluation.]

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5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

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 < 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests.

2. air lock testing acceptance criteria are:

i. overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.

ii. for each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ 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.

5.5.17 Battery Monitoring and Maintenance Program

Brackets are added

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

[This program and the corresponding requirements in LCO 3.8.4, LCO 3.8.5, and LCO 3.8.6 require providing the information and verifications requested in the Notice of Availability for TSTF-500, Revision 2, "DC Electrical Rewrite - Update to TSTF-360," (76FR54510).]

This Program provides controls for battery restoration and maintenance. The program shall be in accordance with IEEE Standard (Std) 450-2002, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications,” as endorsed by NRC RG 1.129, Revision 2 (RG), with RG exceptions and program provisions as identified below:

a. The program allows the following RG 1.129, Revision 2 exceptions:

1. battery temperature correction may be performed before or after conducting discharge tests.

2. RG 1.129, Regulatory Position 1, Subsection 2, “References,” is not applicable to this program.

5.5 Programs and Manuals

5.5.19 Setpoint Control Program

This program shall establish the requirements for ensuring that setpoints for automatic protective devices are initially within and remain within the assumptions of the applicable safety analyses, provides a means for processing changes to instrumentation setpoints, and identifies setpoint methodologies to ensure instrumentation will function as required. The program shall ensure that testing of automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A) verifies that instrumentation will function as required.

a. The program shall list the Functions in the following specifications to which it applies:

1. LCO 3.3.1, “Reactor Protection System (RPS) Instrumentation – Operating”;
2. LCO 3.3.2, “Reactor Protection System (RPS) Instrumentation – Shutdown”;
3. LCO ~~[~~3.3.3, “Control Element Assembly Calculators (CEACs)”~~]~~;

Brackets are deleted

4. LCO 3.3.5, “Engineered Safety Features Actuation System (ESFAS) Instrumentation”;
5. LCO 3.3.7, “Emergency Diesel Generator (EDG) – Loss of Voltage Start (LOVS)”;
6. LCO 3.3.8, “Containment Purge Isolation Actuation Signal (CPIAS)”;
7. LCO 3.3.9, “Control Room Emergency Ventilation Actuation Signal (CREVAS)”;
8. LCO 3.3.10, “Fuel Handling Area Emergency Ventilation Actuation Signal (FHEVAS)”; and
9. LCO 3.3.13, “Logarithmic Power Monitoring Channels”.

5.5 Programs and Manuals

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5.5.19 Setpoint Control Program (continued)

b. The program shall require the nominal trip setpoint (NTSP), allowable value (AV), as-found tolerance (AFT), and as-left tolerance (ALT) (as applicable) of the Functions described in paragraph a. are calculated using the NRC approved setpoint methodology, as listed below. In addition, the program shall contain the value of the NTSP, AV, AFT, and ALT (as applicable) for each Function described in paragraph a. and shall identify the setpoint methodology used to calculate these values:

- APR1400-F-C-NR-14001-P, Rev. 1, "CPC Setpoint Analysis Methodology for APR1400," February 2017,
- APR1400-Z-J-NR-14004-P, Rev. 1, "Uncertainty Methodology and Application of Instrumentation," February 2017, and
- APR1400-Z-J-NR-14005-P, Rev. 1, "Setpoint Methodology for Plant Protection System," February 2017.

Brackets are added

------------------------------------REVIEWER'S NOTE---------------------------------------
[List the NRC safety evaluation report by letter, date, and ADAMS accession number (if available) that approved the setpoint methodologies.]
----

c. The program shall establish methods to ensure that Functions described in paragraph a. will function as required by verifying the as-left and as-found settings are consistent with those established by the setpoint methodology.

Brackets are added

------------------------------------REVIEWER'S NOTE---------------------------------------
[A license amendment request to implement a Setpoint Control Program must list the instrument functions to which the program requirements of paragraph d. will be applied. Paragraph d. shall apply to all Functions in the Reactor Protection System and Engineered Safety Feature Actuation System specifications unless one or more of the following exclusions apply:

1. Manual actuation circuits, automatic actuation logic circuits or to instrument functions that derive input from contacts which have no associated sensor or adjustable device, e.g., limit switches, breaker position switches, manual actuation switches, float switches, proximity detectors, etc. are excluded. In addition, those permissives and interlocks that derive input from a sensor or adjustable device that is tested as part of another TS function are excluded.

5.5 Programs and Manuals

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5.5.19 Setpoint Control Program (continued)

2. Settings associated with safety relief valves are excluded. The performance of these components is already controlled (i.e., trended with as-left and as-found limits) under the ASME Code for Operation and Maintenance of Nuclear Power Plants testing program.

3. Functions and Surveillance Requirements which test only digital components are normally excluded. There is no expected change in result between SR performances for these components. Where separate as-left and as-found tolerance is established for digital component SRs, the requirements would apply.]

Brackets are added

d. The program shall identify the Functions described in paragraph a. that are automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A). These Functions shall be demonstrated to be functioning as required by applying the following requirements during CHANNEL CALIBRATIONS and CHANNEL FUNCTIONAL TESTS that verify the NTSP.

1. The as-found value of the instrument channel trip setting shall be compared with the previous as-left value or the specified NTSP.

2. If the as-found value of the instrument channel trip setting differs from the previous as-left value or the specified NTSP by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel shall be evaluated before declaring the SR met and returning the instrument channel to service. This condition shall be entered in the plant corrective action program.

3. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the instrument channel shall be immediately declared inoperable.

5.6 Reporting Requirements

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5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

3.4.3, "RCS Pressure and Temperature (P/T) Limits";

3.4.6, "RCS Loops – MODE 4";

3.4.7, "RCS Loops – MODE 5 (Loops Filled)";

3.4.10, "Pressurizer Pilot Operated Safety Relief Valves (POSRVs)"; and

3.4.11, "Low Temperature Overpressure Protection (LTOP) System".

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

APR1400-Z-M-NR-14008-P, "Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown."

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

Brackets are added

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

[The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).

2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.

3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC approved methodologies may be included in the PTLR.

5.6 Reporting Requirements

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5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.

5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.

6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.

7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value (2σ) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in $RT_{NDT} + 2\sigma$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.]

Brackets are added

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5.6 Reporting Requirements

5.6.5 Accident Monitoring Report

of

When a report is required by Condition B or F LCO 3.3.11, "Accident Monitoring Instrumentation (AMI)," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.6 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.7 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.9, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced degradation,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. [Repair method utilized and number of tubes repaired by each repair method.]

~~Brackets are deleted~~

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BASES

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LCO 3.0.9

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

Brackets are added

[Adoption of LCO 3.0.9 requires the licensee to make the following commitments:

1. ~~[LICENSEE]~~ commits to the guidance of NUMARC 93-01, Revision 3, Section 11, which provides guidance and details on the assessment and management of risk during maintenance.

Brackets are deleted

2. ~~[LICENSEE]~~ commits to the guidance of NEI 04-08, "Allowance for Non Technical Specification Barrier Degradation on Supported System OPERABILITY (TSTF-427) Industry Implementation Guidance," March 2006.]

Brackets are deleted

LCO 3.0.9 establishes conditions under which systems described in the TS are considered to remain OPERABLE when required barriers are not capable of providing their related support function(s).

Barriers are doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in TS, that support the performance of the safety function of systems described in the TS. This LCO states that the supported system is not considered to be inoperable solely due to required barriers not capable of performing their related support function(s) under the described conditions. LCO 3.0.9 allows 30 days before declaring the supported system(s) inoperable and the LCO(s) associated with the supported system(s) not met. A maximum time is placed on each use of this allowance to ensure that as required barriers are found or are otherwise made unavailable, they are restored.

However, the allowable duration may be less than the specified maximum time based on the risk assessment.

If the allowed time expires and the barriers are unable to perform their related support function(s), the supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

This provision does not apply to barriers which support ventilation systems or to fire barriers. The TS for ventilation systems provide specific Conditions for inoperable barriers. Fire barriers are addressed by other regulatory requirements and associated plant programs. This provision does not apply to barriers which are not required to support system OPERABILITY (see NRC Regulatory Issue Summary 2001-09, "Control of Hazard Barriers," dated April 2, 2001).

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BASES

BACKGROUND (continued)

The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants.

The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The CRHS operation to maintain the control room temperature is discussed in FSAR, Subsection 9.4.1 (Ref. 1). Upon receipt of the actuating signal(s), normal makeup air supply to the AHU is isolated, and the stream of ventilation air is recirculated through the filter trains of the CREACS.

Brackets are added

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

[The need for toxic gas isolation mode will be determined by the COL applicant.]

Actuation of the CRHS places the system into the emergency mode for protection from radiation [or the toxic gas isolation mode for protection from toxic gas, depending on the initiation signal]. Upon receipt of actuation signal of the emergency mode of operation, the unfiltered normal makeup air path is isolated, exhaust isolation dampers are closed, and CREACS of the operating division is automatically started. The emergency mode initiates pressurization and filtered ventilation of the air supply to the CRE.

Outside air is filtered, and then added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary.

[Upon detection of a toxic gas, the toxic gas detector will initiate complete closure of outside intake isolation dampers to the CRE.]

The air entering the CRE is continuously monitored by radiation [and toxic gas] detectors. One detector output above the setpoint causes actuation of the emergency mode [or the toxic gas isolation mode] as required. [The actions of the toxic gas isolation mode take precedence, and will override the action of the emergency mode.]

The CRHS operating at a flow rate of 6,286 cmh (3,700 cfm) pressurizes the control room to about 3.175 mm (0.125 in) water gauge relative to external areas adjacent to the CRE boundary. The CRHS operation in maintaining the CRE habitable is discussed in FSAR, Section 6.4 (Ref. 2).

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BASES

ACTIONS (continued)

this time period and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that could adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan, and possibly repair and test most problems with the CRE boundary.

D.1 and D.2

In MODE 1, 2, 3, or 4, if the inoperable CREACS or CRSRS division or the CRE boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and 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 unit systems.

E.1 and E.2

Required Action E.1 is performed manually.

[In MODE 5 or 6, or] [During] movement of irradiated fuel assemblies, if Required Action A.1 or B.1 cannot be completed within the required Completion Time, the CREACS and CRSRS of the OPERABLE CRHS division must be immediately placed in the emergency MODE of operation. This action ensures that the remaining division is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action E.1 is Required Action[s] E.2[.1 and E.2.2] to immediately suspend activities that could result in a release of radioactivity that may require isolation of CRE. This places the unit in a condition that minimizes the accident risk.

This does not preclude the movement of fuel assemblies to a safe position.

Brackets are added

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

[The need for toxic gas isolation mode will be determined by the COL applicant.]

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BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the ABCAEES is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS.

In MODES 5 and 6, the ABCAEES is not required to be OPERABLE, since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one ABCAEES division inoperable, the inoperable ABCAEES division must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE division is adequate to perform the ABCAEES function.

The 7 day Completion Time is appropriate because the risk contribution of the system is less than that for the ECCS (72 hour Completion Time) and this system is not a direct support system for the ECCS. The 7 day Completion Time is reasonable, based on the low probability of a DBA occurring during this time period and the consideration that the remaining division can provide the required capability.

B.1

Brackets are added

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

[Adoption of Condition B is dependent on a commitment from the operating licensee to have guidance available describing compensatory measures to be taken in the event of an intentional and unintentional entry into Condition B.]

Brackets are added

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

[The need for toxic gas isolation mode will be determined by the COL applicant.]

If the mechanical penetration room or safety-related mechanical equipment room boundary is inoperable, the ABCAEES divisions cannot perform their intended functions. Actions must be taken to restore an OPERABLE mechanical penetration room and safety-related mechanical equipment room boundary within 24 hours. During the period that the mechanical penetration room or safety-related mechanical equipment room boundary is inoperable, appropriate compensatory measures [consistent with the intent, as applicable, of GDC 19, 60, 64 and 10 CFR Part 100] should be utilized to protect plant personnel from potential hazards such as radioactive contamination, [toxic gases], smoke, temperature, and physical security. Preplanned measures should be available to address these concerns for intentional and

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BASES

SURVEILLANCE REQUIREMENTS (continued)

These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ 0.9 results in voltages on the emergency busses that are too high. Under these conditions, the power factor should be maintained as close as practicable to 0.9 while still maintaining acceptable voltage limits on the emergency busses. In other circumstances, the grid voltage could be such that the EDG excitation levels needed to obtain a power factor of 0.9 may not cause unacceptable voltages on the emergency busses, but the excitation levels are in excess of those recommended for the EDG. In such cases, the power factor shall be maintained as close as practicable to 0.9 without exceeding the EDG excitation limits.

Brackets are added

REVIEWER'S NOTE

[The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:

a. Performance of the SR will not render any safety system or component inoperable.
b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems.
c. Performance of the SR or failure of the SR will not cause or result in an AOO with attendant challenge to plant safety systems.]

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BASES

SURVEILLANCE REQUIREMENTS (continued)

This power factor is representative of the actual inductive loading an EDG would see under design basis accident conditions. Under certain conditions, however, Note 2 allows the Surveillance to be conducted at a power factor other than ≤ 0.9. These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ 0.9 results in voltages on the emergency busses that are too high. Under these conditions, the power factor should be maintained as close as practicable to 0.9 while still maintaining acceptable voltage limits on the emergency busses. In other circumstances, the grid voltage could be such that the EDG excitation levels needed to obtain a power factor of 0.9 may not cause unacceptable voltages on the emergency busses, but the excitation levels are in excess of those recommended for the EDG. In such cases, the power factor shall be maintained as close as practicable to 0.9 without exceeding the EDG excitation limits.

Brackets are added

REVIEWER'S NOTE

[The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:

a. Performance of the SR will not render any safety system or component inoperable.

b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems.

c. Performance of the SR or failure of the SR will not cause or result in an AOO with attendant challenge to plant safety systems.]

BASES

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SURVEILLANCE REQUIREMENTS (continued)

Brackets are added

REVIEWER'S NOTE

[The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:

a. Performance of the SR will not render any safety system or component inoperable.

b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems.

c. Performance of the SR or failure of the SR will not cause or result in an AOO with attendant challenge to plant safety systems.]

SR 3.8.1.14

NRC RG 1.9 (Ref. 3) requires demonstration that the EDGs can start and run continuously at full load capability for an interval of not < 24 hours, ≥ 2 hours of which is at a load equivalent to 110% of the continuous duty rating and the remainder of the time at a load equivalent to the continuous duty rating of the EDG. The EDG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

The load band is provided to avoid routine overloading of the EDG. Routine overloading could result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY.

The 18 month Frequency is consistent with the recommendations of NRC RG 1.9 (Ref. 3) takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

The 18 month Frequency is consistent with the recommendations of NRC RG 1.9 (Ref. 3), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, 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 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 Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the 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 and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

Brackets are added

REVIEWER'S NOTE

[The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:

a. Performance of the SR will not render any safety system or component inoperable.

b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems.

c. Performance of the SR or failure of the SR will not cause or result in an AOO with attendant challenge to plant safety systems.]

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B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within the containment will be restricted from leakage to the environment when the Limiting Condition for Operation (LCO) requirements are met. In MODES 1, 2, 3, and 4 this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1 "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not present, therefore, less stringent requirements are needed to isolate the containment from the outside atmosphere. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the ANSI/ANS 56.8-1994 leakage criteria and tests are not required.

The containment structure serves to contain fission product radioactivity which could be released from the reactor core following a Design Basis Accident (DBA), such that offsite radiation exposures are maintained well within the requirements of 10 CFR Part 50.34. Additionally, this structure provides radiation shielding from the fission products which could be present in the containment atmosphere following accident conditions.

Brackets are added

-----------------------------------Reviewer's Note-----------------------------------

[The number of bolts, material, size, and analysis supporting the hatch capability to support the dead weight (at a minimum) will be determined by the COL applicant.]

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least [four bolts]. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment airlocks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 operation in accordance with LCO 3.6.2, "Containment Airlocks." Each airlock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required.