PMComanchePekNPEm Resource

From: Monarque, Stephen
Sent: Monday, November 02, 2009 5:01 PM
To: ComanchePeakCOL Resource
Subject: FW: COLA Parts 4, 10, and 11 Update Tracking Report- nonpublic as SUNSI needs to be done
Attachments: COLA Part 10.pdf; COLA Part 04.pdf; TXNB-09053 UTR COLA Parts 4, 10, and 11.pdf

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Subject: COLA Parts 4, 10, and 11 Update Tracking Report

Luminant has submitted the attached Update Tracking Report for COLA Parts 4, 10, and 11 to the NRC. Part 4 is Tech Specs; Part 10 is ITAAC/License Conditions; and Part 11 is the QAPD. The QAPD was 28 Mb and is not included with this message. Stephen Monarque and the Document Control Desk each received a CD with all attachments. If there are any questions regarding the UTR, please contact me or contact Don Woodlan (Donald.Woodlan@luminant.com, 254-897-6887).

Thanks,

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October 16 2009

Comanche Peak Nuclear Power Plant, Units 3 & 4
COL Application

Part 10

ITAAC and Proposed License Conditions

Update Tracking Report

Revision 0

Revision History

Revision	Date	Update Description
0	10/16/2009	Original Issue Updated Sections: 1, 3, Appendix

Tracking Report Revision List

Change ID No.	Section	Page from Part 10 Rev.0	Reason for change	Change Summary	Rev. of T/R
RCOL2_05.03.02-3	2	3, 4	RAI No. 8 Response Luminant letter TXNB-09-028 Date 08/7/09	Added specific license condition	-
DCD_14.03.06-15	1 Appendix A.4	3 24	Add ITAAC	Added Offsite Power System ITAAC	0
DCD_13.06-21	1 Appendix C Table C-1 (Sheet 5 of 5)	3 41	Add ITAAC	Added Plant Specific Security Hardware ITAAC	0
CTS-00841	2.3, 2.4, 3,	3, 4	Proposed NRC Generic Combined License	License Conditions contained in Proposed Generic Combined License	0

1. ITAAC

The ITAAC for the COLA consist of the following:

1) Design Certification ITAAC are contained in DCD Tier 1 and are incorporated by reference.
2) Plant-Specific ITAAC are provided in Appendices A.1, A.2, ~~and~~ A.3, and A.4. The design description information contained in the Appendices is a compilation of information from various sources in the FSAR and is included to assist the reader in reviewing information pertinent to the Plant-Specific ITAAC. DCD_14.03.06-15
3) Emergency Planning ITAAC are provided in Appendix B.
4) Physical Security ITAAC for the DCD are contained in DCD Tier 1 and are incorporated by reference. Plant Specific Security ITAAC are provided in Appendix C. DCD_13.06-21

2. Proposed License Conditions

The NRC and industry are currently evaluating the appropriate license conditions for a Combined Operating License (COL). Identified below are several possible topics for license conditions that serve as a starting point for consideration. The listing is not final nor are all items necessarily appropriate. As a result, this section will not be updated during the COL review until further NRC and industry guidance is available. As specific license conditions are identified they will be added to section 3 below. RCOL2_05.03.02-3

2.1 Completion of ITAAC

Completion of the ITAAC listed in the previous section may be a proposed license condition to be satisfied prior to fuel load. However, this license condition may not be necessary as the ITAAC may be adequately controlled by the regulations.

2.2 COL Holder Items

COL Information Items are identified in Chapter 1 of the FSAR (Table 1.8-201) and are cross-referenced to identify the section in this COLA that addresses each Information Item from the referenced certified design. Items that cannot be resolved prior to issuance of the COL are identified as Holder Items. Implementation of all Holder Items by the milestone stated in the relevant section of the FSAR, is potential condition to the license. There are alternate methods to track these items including a commitment tracking system or NRC inspection schedules. If such alternate systems are found to be appropriate, a license condition may not be necessary or a more limited license condition addressing only selected Holder Items may be appropriate.

2.3 Operational Programs

Operational Programs are identified in Table 13.4-201 and their implementation by the milestones indicated in the Table is a potential condition to the license. Some of these programs may be adequately controlled by other methods such as the regulations, the technical specifications or a commitment tracking system and will not need to be addressed in a license condition. A proposed license condition is provided in section 3 below based upon the current information in Chapter 13 of the COLA FSAR. CTS-00841

2.4 Environmental Protection Plan

The Environmental Protection Plan (EPP) and its implementation may also be a potential condition to the license. The EPP has typically been an appendix to the operating license and that precedent may be followed for COLs as well. No plant specific environmental items have been identified which are not adequately controlled by regulations, the appropriate permits, etc. and thus an EPP has not been proposed and is not needed. CTS-00841

2.5 Technical Specifications

Implementation of Technical Specifications prior to fuel load could also constitute a potential condition to the license. The Technical Specifications have typically been an appendix to the operating license and that precedent may be followed for COLs as well.

2.6 Others

The current operating licenses have some typical license conditions in areas such as security, fire protection and others. These current license conditions may or may not apply to COLs.

3. Specific Proposed License Conditions

The ~~only~~ license conditions identified thus far during the COL development and review are ~~is~~:

CTS-00841

Proposed License Condition	**Source**
The plant-specific PTS evaluation of the as-procured reactor vessel material properties will be submitted to the NRC within 12 months following acceptance of the reactor vessel.	Answer to RAI 2353 (CP RAI #8) question 05.03.02-3 as provided in TXNB-09028 dated August 7, 2009.
The licensee shall implement the programs or portions of programs identified in the table below on or before the associated milestones.	COLA FSAR Table 13.4-201 Items 3, 5, 6, 8, 9, 10, 12, 15, 18, and 19.

RCOL2_05.03.02-3

Operational Programs to be implemented per License Condition above: CTS-00841

Program Title	Milestone
Environmental Qualification Program	Prior to Initial Fuel Load
Reactor Vessel Material Surveillance Program	Prior to Initial Criticality
Preservice Testing Program	Prior to Initial Fuel Load
Fire Protection Program	Prior to fuel receipt for elements of the Fire Protection Program necessary to support receipt and storage of fuel on-site. Prior to initial fuel load for elements or the Fire Protection Program necessary to support fuel load and plant operation.
Process and Effluent Monitoring and Sampling Program – Radiological Effluent Technical Specifications/Standard Radiological Effluent Controls	Prior to receipt of radioactive material on-site
Process and Effluent Monitoring and Sampling Program – Offsite Dose Calculation Manual	Prior to receipt of radioactive material on-site
Process and Effluent Monitoring and Sampling Program – Radiological Environmental Monitoring Program	Prior to receipt of radioactive material on-site
Process and Effluent Monitoring and Sampling Program – Process Control Program	Prior to receipt of radioactive material on-site
Radiation Protection Program	Prior to initial receipt of by-product, source, or special nuclear materials (excluding Exempt Qualities as described in 10 CFR 30.18) for those elements of the Radiation Protection (RP) Program necessary to support such receipt Prior to fuel receipt for those elements of the RP Program necessary to support receipt

Program Title	Milestone
	and storage of fuel on-site. Prior to fuel load for those elements of the RP Program necessary to support fuel load and plant operation Prior to first shipment of radioactive waste for those elements of the RP Program necessary to support shipment of radioactive waste.
Reactor Operator Training Program	18 months prior to scheduled fuel load.
Security Program – Physical Security Program	Prior to receipt of fuel on site.
Security Program- Safeguards Contingency Program	Prior to receipt of fuel on site.
Security Program – Training and Qualification Program	Prior to receipt of fuel on site.
Motor-Operated Valve Testing	Prior to initial fuel load.
Initial Test Program	Prior to the first construction test for the Construction Test Program. Prior to the first preoperational test for the Preoperational Test Program. Prior to initial fuel loading for the Startup Test Program.
Fitness for Duty Program – Construction Mgt & Oversight personnel	Prior to on site construction of safety or security related SSCs.
Fitness for Duty Program – Construction Workers & first Line Supv.	Prior to on site construction of safety or security related SSCs.
Fitness for Duty Program – Operations Phase Program	Prior to fuel receipt

CTS-00841

PART 10 - APPENDIX A.4

DCD_14.03.06-15

OFFSITE POWER SYSTEM (PORTIONS OUTSIDE THE SCOPE OF THE CERTIFIED DESIGN)

A.4.1 Inspections, Tests, Analysis, and Acceptance Criteria

Table A.4-1 describes the inspections, tests, analyses, and associated acceptance criteria for the Offsite power system portions outside the scope of the certified design.

Part 10 - ITAAC and Proposed License Conditions

Appendix A.4

Table A.4-1 (Sheet 1 of 2)

DCD_14.03.06-15

Offsite Power System (Portions Outside the Scope of the Certified Design) Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The electrical system has a minimum of two independent offsite transmission circuits from the transmission network (TN) to the safety buses with no intervening non-safety buses (direct connection).	1. Inspection of the as-built transmission circuits will be performed.	1. The as-built electrical system has two independent offsite transmission circuits from the TN to the safety buses with no intervening non-safety buses (direct connection).
2. The offsite TN, during steady-state operation, does not cause voltage variations beyond an acceptable tolerance of the loads' nominal ratings.	2. Analyses of the as-built offsite TN voltage variability and steady state load requirements will be performed.	2. A report exists and concludes that the as-built offsite TN, during steady state operation, does not cause voltage variations beyond design limits.
3. The offsite TN normal steady state frequency is within an acceptable tolerance of 60Hz during recoverable periods of system instability.	3. Analyses of the as-built offsite TN normal steady state frequency will be performed.	3. A report exists and concludes that the as-built TN normal steady state frequency is within design frequency limits during recoverable periods of instability.
4. The offsite transmission circuits have the capacity and capability to power the required loads during steady state, transient, and postulated events and accident conditions.	4. Analyses of the as-built offsite transmission circuits from the TN to the safety buses will be performed.	4. A report exists and concludes that the as-built offsite transmission circuits have the capacity and capability to power the required loads during steady state, transient, and postulated events and accident conditions.
5.a Independence between the offsite circuits and the onsite Class 1E electrical system and components is maintained.	5.a Tests and analyses on the as-built offsite circuits and onsite class 1E electrical system and components will be performed.	5.a The offsite circuits are isolated from the onsite Class 1E electrical system and components.
5.b The offsite circuits are physically separated from the onsite Class 1E electrical system and components.	5.b Inspections of the as-built offsite circuits and onsite Class 1E electrical system and components will be performed.	5.b The as-built offsite circuits are physically separated from the onsite Class 1E electrical system and components.
6. Lightning protection and grounding features are provided for the offsite circuits from the TN to the safety buses.	6. Inspection of the as-built offsite circuits from the TN to the safety buses will be performed.	6. Lightning protection and grounding features exist for the system and components of the offsite circuits from the TN to the safety buses.

DCD_14.03.06-15

Table A.4-1 (Sheet 2 of 2)

Offsite Power System (Portions Outside the Scope of the Certified Design) Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. MCR alarms and displays for monitoring the switchyard equipment status can be retrieved in the MCR.	7. Inspection will be performed for the retrievability of the as-built switchyard equipment status in the as-built MCR.	7. MCR alarms and displays for monitoring the switchyard equipment status can be retrieved in the as-built MCR.
8. If power through the preferred power supply is not available, the offsite electrical system has the capability to automatic fast transfer to the non-preferred power supply if available.	8. Inspection of the as-built offsite electrical system will be performed.	8. The as-built offsite electrical system is automatically transferred to the non-preferred power supply in power is not available through the preferred power supply.
9. The Switchyard agreement and protocols between the NPP and the TN system operator/owner assess the risk and probability of a loss of offsite power due to performing maintenance activities on the electrical system.	9. Inspection of the switchyard agreement and protocols between the NPP and the TN owner/operator will be performed.	9. The switchyard agreement and protocols between the NPP and the TN owner/operator assess the risk and probability of a loss of offsite power due to performing maintenance activities on the electrical system.
10. The offsite electrical system (switchyard) design assesses the probability of losing electric power as a result of or coincident with, the loss of power generated by the nuclear unit, the loss of power from the TN, or the loss of the largest load.	10. Analyses of the as-built offsite electrical system for transient stability will be performed.	10. A report exists and concludes that the as-built offsite electrical system design assesses the probability of losing electric power as a result of or coincident with the loss of power generated by the nuclear unit, the loss of power from the TN, or the loss of the largest load.

PART 10 - APPENDIX C
PHYSICAL SECURITY HARDWARE

C.1 Inspections, Tests, Analyses, and Acceptance Criteria

DCD_13.06-21

Table C-1 describes the inspections, tests analyses, and associated acceptance criteria for the site-specific physical security hardware.

DCD_13.06-21

Table C-1 Physical Security Hardware Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 1 of 5)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.b Access to vital equipment requires passage through at least two physical barriers.	1.b Inspections will be performed of vital equipment locations.	1.b. Vital equipment is located such that access to the vital equipment requires passage through at least two physical barriers.
2.a Physical barriers for the protected area perimeter are not part of vital area barriers.	2.a Inspections of the protected area perimeter barriers will be performed.	2.a Physical barriers at the perimeter of the protected area are separated from any other barrier designated as a Vital Area barrier.
2.b Penetrations through the protected area barrier must be secured and be capable of being monitored.	2.b Inspections will be performed of penetrations through the protected area barrier.	2.b Penetrations and openings of a passable size through the protected area barrier are secured and monitored by intrusion detection equipment.
2.c Unattended openings of passable size that intersect a security boundary such as underground pathways must be protected by a physical barrier and monitored by intrusion detection equipment.	2.c Inspections will be performed of unattended openings of passable size within the protected area barriers.	2.c Unattended openings of a passable, (such as underground pathways) that intersect a security boundary (such as the protected area barrier), are protected by a physical barrier and monitored by intrusion detection equipment
3.a Isolation zones exist in outdoor areas adjacent to the physical barrier at the perimeter of the protected area that allow sufficient size for observation and assessment on either side of the barrier.	3.a Inspections of the isolation zones outdoor areas adjacent to the physical barrier will be performed.	3.a The isolation zones exist in outdoor areas adjacent to the physical barrier at the perimeter of the protected area and allow 20 feet for observation and assessment of the activities of people on either side of the barrier.
3.b Where permanent buildings do not allow a sufficient distance for observation on the inside of the protected area, the building walls are immediately adjacent to, or an integral part of, the protected area barrier, and the (license applicant specified) observation distance does not apply.	3.b Inspections of the part of the building that constitutes the protected area will be performed.	3.b Where permanent buildings do not allow a 20 feet distance on the inside of the protected area, the building walls are immediately adjacent to, or an integral part of, the protected area barrier and the 20 feet observation distance does not apply.

Table C-1 Physical Security Hardware Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 2 of 5)

DCD_13.06-21

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4.a Intrusion detection system (IDS) can detect penetration or attempted penetration of the protected area perimeter barrier and subsequent alarms annunciate concurrently in at least two continuously manned onsite alarms stations, (central and secondary alarm stations).	4.a Tests, inspections or a combination of tests and inspections of the intrusion detection system will be performed.	4.a The intrusion detection system can detect penetration or attempted penetration of the protected area perimeter barrier and subsequent alarms annunciate concurrently in at least two continuously manned onsite alarms stations, (central and secondary alarm stations).
4.b Video image recording equipment with real-time and play-back capability provides the ability to assess detected assessment activities before and after each alarm annunciation within the isolation zone.	4.b Tests, inspections or a combination of tests and inspections of the video assessment equipment will be performed.	4.b Video image recording equipment with real-time and play-back capability provide the ability to display activities before and after each alarm annunciation within the isolation zone.
4.c Intrusion detection and assessment equipment at the protected area perimeter remains operable from an uninterruptible power supply in the event of the loss of normal power.	4.c Tests, inspections or a combination of tests and inspections of the uninterruptible power supply will be performed.	4.c Intrusion detection and assessment equipment at the protected area perimeter remains operable from an uninterruptible power supply in the event of the loss of normal power.
5. Isolation zones and exterior areas within the protected area are provided with illumination to permit observation of abnormal presence or activity of persons or vehicles.	5. Inspections of the Illumination in isolation zones and exterior areas of the protected will be performed.	5. Illumination in isolation zones and exterior areas within the protected area is 0.2 foot-candles measured horizontally at ground level or, alternatively, sufficient to permit observation.
6.b The external walls, doors, ceiling and floors in the secondary alarm station and the last access control function for access to the protected area are bullet resistant.	6.b Type test, analysis or a combination of type test and analysis of the external walls, doors, ceiling and floors in the secondary alarm station and the last access control function for access to the protected area will be performed.	6.b A report exists and concludes that the external walls, doors, ceilings, floors in the secondary alarm station and the last access control function for access to the protected area are bullet resistant to , UL752 (2006) Level 4.

Table C-1 Physical Security Hardware Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 3 of 5)

DCD_13.06-21

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. The vehicle barrier system is installed and located at the necessary stand-off distance to protect against the DBT vehicle bombs.	7. Inspections will be performed for the vehicle barrier system.	7. The vehicle barrier system will protect against the DBT vehicle bombs based upon the stand-off distance for the system.
8.a Access control points are established to control personnel and vehicle access into the protected area.	8.a Tests, inspections, or combination of tests and inspections of installed systems and equipment will be performed.	8.a Access control points exist for the protected area and are configured to control access.
8.b Access control points are established to detect firearms, explosives, and incendiary devices at the protected area personnel access points.	8.b Tests, inspections, or combination of tests and inspections of installed systems and equipment will be performed.	8.b The detection equipment at the protected area personnel access points is capable of detecting firearms, explosives, and incendiary devices.
9. A security access control system with numbered picture badges is installed for use by individuals who are authorized access to protected areas without escort.	9. Tests of the access control system with numbered picture badges will be performed.	9. The access authorization system utilizes numbered picture badges, and authorizes protected area access only to those personnel with unescorted access authorization.
10.b Unoccupied vital areas are locked and alarmed with activated intrusion detection systems that annunciate in the secondary alarm station.	10.b Tests, inspections, or a combination of tests and inspections of unoccupied vital areas intrusion detection equipment and locking devices will be performed.	10.b Unoccupied vital areas are locked and intrusion is detected and annunciated in the secondary alarm station.
11.a-2 Security alarm annunciation and video assessment information are available concurrently in the secondary alarm station.	11.a-2 Tests, inspections or a combination of tests and inspections of alarm annunciation and video assessment equipment will be performed.	11.a-2 Security alarm annunciation and video assessment equipment information is available concurrently in the secondary alarm station.
11.b-2 The secondary alarm station is located inside a protected area and the interior of the secondary alarm station is not visible from the perimeter of the protected area	11.b-2 Inspections of the secondary alarm station locations will be performed.	11.b-2 The secondary alarm station is located inside a protected area and the interior of the secondary alarm station is not visible from the perimeter of the protected area.

Table C-1 Physical Security Hardware Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 4 of 5)

DCD_13.06-21

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11.c Central and secondary alarm stations are designed and equipped such that, in the event of a single act, in accordance with the design basis threat of radiological sabotage, the design enables the survivability of equipment needed to maintain the functional capability of either alarm station to: (1) detect and assess alarms (2) initiate and coordinate an adequate response to alarms (3) summon offsite assistance, and (4) provide effective command and control.	11.c Tests, inspections or a combination of tests and inspections of the central and secondary alarm stations will be performed.	11.c Central and secondary alarm stations are designed, equipped and constructed such that , in the event of a single act, in accordance with the design basis threat of radiological sabotage, the design enables the survivability of equipment needed to maintain the functional capability of either alarm station to: (1) detect and assess alarms (2) initiate and coordinate an adequate response to alarms (3) summon offsite assistance, and (4) provide effective command and control.
11.d Both the central and secondary alarm stations are constructed, protected, and equipped to the standards for the central alarm station (stations need not be identical in design).	11.d Tests, inspections or a combination of tests and inspections of the central and secondary alarm stations will be performed.	11.d The central alarm station and secondary alarm station are constructed, protected, and equipped to the same standards for functional redundancy (stations need not be identical in design).
13.b-2 Intrusion detection and assessment systems are designed to provide visual display and audible annunciation in the secondary alarm station.	13.b-2 Tests will be performed on Intrusion detection and assessment systems.	13.b-2 The intrusion detection system provides a visual display and audible annunciation of alarms in the secondary alarm station.
15.b Emergency exits through the protected area perimeter are alarmed and secured by locking devices that allow prompt egress during an emergency.	15.b Tests, inspections or a combination of tests and inspections of emergency exits through the protected area perimeter will be performed.	15.b Emergency exits through the protected area perimeter are alarmed and secured by locking devices that allow prompt egress during an emergency.

Table C-1 Physical Security Hardware Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 5 of 5)

DCD_13.06-21

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
16.a-2 The secondary alarm station has conventional (land line) telephone service with local law enforcement authorities and a system for communication with the main control room.	16.a-2 Tests, inspections, or a combination of tests and inspections of the secondary alarm station communications capability with local law enforcement authorities and main control room will be performed	16.a-2 The secondary alarm station is equipped with conventional (land line) telephone service with local law enforcement authorities and has a system for continuous communication with the main control room.
16.b-2 The secondary alarm station is capable of continuous communication with security personnel.	16.b-2 Tests, inspections, or a combination of tests and inspections of the secondary alarm station continuous communication capabilities will be performed.	16.b-2 The secondary alarm station is capable of continuous communication with security officers, watchmen or armed response individuals, or other security personnel that have responsibilities during a contingency event.

October 16 2009

Comanche Peak Nuclear Power Plant, Units 3 & 4
COL Application

Part 4

Technical Specifications

Update Tracking Report

Revision 1

Revision History

Revision	Date	Update Description
0	9/11/2009	Updated Chapters: Specifications, Bases
1	10/16/2009	Updated Chapters: Specifications, Bases

Introduction

Introduction – Tracking Report Revision List

Change ID No.	Section	Page	Reason for change	Change Summary	Rev. of T/R

Specifications

Specifications – Tracking Report Revision List

Change ID No.	Section	Page	Reason for change	Change Summary	Rev. of T/R
-	-	-	Incorporate the DCD Chapter 16 changes that are relevant to Part 4 changes.	Incorporate changes as describe in MHI Letter UTR Rev.0 # UAP-HF-09081 dated 03/06/2009 UTR Rev.1 # UAP-HF-09222 dated 04/30/2009 UTR Rev. 3 # UAP-HF-09413 dated 08/03/2009	0
-	-	-	Incorporate the DCD Chapter 16 changes that are relevant to Part 4 changes.	Incorporate changes as describe in MHI Letter UTR Rev.4 # UAP-HF-09454 dated 18/09/2009	1

1.1 Definitions

CHANNEL OPERATIONAL TEST (COT) - Analog (application of test for analog equipment)	~~For analog equipment, a~~A COT - Analog shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT - Analog shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT - Analog may be performed by means of any series of sequential, overlapping, or total channel steps.
CHANNEL OPERATIONAL TEST (COT) (application of test for digital equipment, PSMS)	~~For the PSMS,~~A COT is a check of the PSMS software memory integrity to ensure there is no change to the internal PSMS software that would impact its functional operation~~, including digital Trip Setpoint values~~ or the continuous self-test function. The PSMS is self-tested on a continuous basis from the digital side of all input modules to the digital side of all output modules. Self-testing also encompasses all ~~digital Trip Setpoints and trip functions. Digital Trip Setpoints are maintained in non-volatile software memory within each PSMS train~~data communications within a PSMS train, between PSMS trains and between the PSMS and PCMS. For the PSMS ~~T~~the self-testing is described in Topical Report, “Safety I&C System Description and Design Process,“ MUAP-07004 Section 4.3 and Topical Report, “Safety System Digital Platform -MELTAC-,“ MUAP-07005 Section 4.1.5. The software memory integrity test is described in Topical Report, “Safety I&C System Description and Design Process,“ MUAP-07004 Section 4.4.1 and Topical Report, “Safety System Digital Platform -MELTAC-,“ MUAP-07005 Section 4.1.4.1.c.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit-specific document that provides cycle-specific parameter limits. These cycle-specific parameter limits shall be determined for each cycle in accordance with Specification 5.6.3. Plant operation within these limits is addressed in individual Specifications.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
S. Required Action and associated Completion Time for Condition N, Q, or R not met.	S.1 Be in MODE 3.	6 hours
T. Main Turbine Stop Valve Position channel inoperable	-------------------NOTE------------------ One channel may be bypassed for up to 12 hours for surveillance testing. --- T.1 Place channel in trip. T.2 Reduce thermal power to < P-7	 12 hours 18 hours

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.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.2	----------------------------NOTE~~S~~------------------------- 1. Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. ~~2. Adjust nuclear instrument channel if absolute difference is > 1%.~~ -- Compare results of calorimetric heat balance calculation to power range channel output. Adjust power range channel output if calorimetric heat balance calculations results exceed power range channel output by more than +2% RTP.	In accordance with the Surveillance Frequency Control Program

Table 3.3.1-1 (page 5 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
14. ECCS ~~a~~Actuation	1,2	3 trains	M	SR 3.3.1.5	NA	NA
15. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2(d)	2	O	SR 3.3.1.7 SR 3.3.1.10	±5% of span	1E-10 A
b. Low Power Reactor Trips Block, P-7	1	1 per train	P	SR 3.3.1.5	NA	NA
c. Power Range Neutron Flux, P-10	1,2	4	O	SR 3.3.1.7 SR 3.3.1.10	±4% RTP	10%RTP
d. Turbine Inlet Pressure, P-13	1	~~4~~3	P	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9	±2.5% of span	10%~~RTP~~ Turbine Power
16. Reactor Trip Breakers (RTBs)	1,2	3 trains(i)	N,S	SR 3.3.1.4 SR 3.3.1.13	NA	NA
	3(b), 4(b), 5(b)	3 trains(i)	D	SR 3.3.1.4 SR 3.3.1.13	NA	NA

(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

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

(i) Two reactor trip breakers per train.

Table 3.3.2-1 (page 6 of 9)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6. Emergency Feedwater Actuation						
a. Manual Initiation	1,2,3	3 trains	F	SR 3.3.2.6	NA	NA
b. Actuation Logic and Actuation Outputs	1,2,3	3 trains	J,T	SR 3.3.2.2 SR 3.3.2.4	NA	NA
c. Low SG Water Level	1,2,3	3 per SG	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.7 SR 3.3.2.8	±3% of span	13% of span
d. ECCS Actuation	Refer to Function 1 (ECCS Actuation) for all initiation functions and requirements.					
e. LOOP Signal	1,2,3	3 per bus for each EFW train	F	SR 3.3.2.5 SR 3.3.2.7 SR 3.3.2.8	~~±1.5% of span~~≥ 4830 V with a time delay of ≤ 2 second	~~4727~~4934 V(l) with ≤ 2 sec time delay
f. Trip of all Main Feedwater Pumps	1,2	1 per pump	H	SR 3.3.2.6 SR 3.3.2.8	NA	NA

(l) Nominal Trip Setpoint

3.3 INSTRUMENTATION

3.3.3 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3 The PAM instrumentation for each Function in Table 3.3.3-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

------------------------------NOTE~~S~~------------------------------

~~1.LCO 3.0.4 not applicable~~

~~2~~1.Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1	Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1	---------------NOTE--------------- 1. For RCS Hot and Cold Leg Temperatures, this Condition is applicable only if at least one channel (Hot or Cold) is operable in each loop. Otherwise, go to Condition C. 2. For SG Water Level and EFW flow, this condition is applicable only if at least one channel (Level or flow) is operable in each loop. Otherwise, go to Condition C. ------------------------------------- Initiate action in accordance with Specification 5.6.5.	Immediately

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.18 RCS Loops – Test Exceptions

LCO 3.4.18 The requirements of LCO 3.4.4, "RCS Loops – MODES 1 and 2," may be suspended with THERMAL POWER < P-7.

APPLICABILITY: MODES 1 and 2 during startup and PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER ≥ P-7.	A.1 Open reactor trip breakers.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.18.1	Verify THERMAL POWER is < P-7.	1 hour
SR 3.4.18.2	Perform a COT for each power range neutron flux - low channel, intermediate range neutron flux channel, P-10, and P-13.	Prior to initiation of startup and PHYSICS TESTS
SR 3.4.18.3	Perform an ACTUATION LOGIC TEST on P-7.	Prior to initiation of startup and PHYSICS TESTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	-----------------------------NOTE---------------------------- Only required to be performed when containment air temperature is < 32°F or > 120°F. --- Verify RWSP borated water temperature is ≥ 32°F and ≤ 120°F.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.2	Verify RWSP borated water volume is ≥ ~~329,150~~583,340 gallons.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.3	Verify RWSP boron concentration is ≥ 4000 ppm and ≤ 4200 ppm.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Operate each Annulus Emergency Exhaust System train for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.7.11.2	Perform required Annulus Emergency Exhaust System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.11.3	Verify each Annulus Emergency Exhaust System train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.11.4	Verify the associated room can be maintained at ~~one Annulus Emergency Exhaust System train can maintain~~ a pressure ≤ -0.25 inches water gauge relative to atmospheric pressure using one Annulus Emergency Exhaust System train during the accident condition at a flow rate of ≤ 5600 cfm within 240 seconds after a start signal.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.8 Decay Time

LCO 3.9.8 The reactor shall be subcritical for ≥ 24 hours.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor subcritical < 24 hours.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.8.1	Verify that the reactor has been subcritical for ≥ 24 hours by verification of the date and time of subcriticality.	Prior to movement of irradiated fuel assemblies within reactor vessel.

5.2 ORGANIZATION

5.2.2 Unit Staff (continued)

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.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. A radiation protection technician and chemistry technician shall be on site when fuel is in the reactor. The positions 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 positions.

d. ~~Administrative procedures shall be developed and implemented to limit the working hours of personnel who perform safety related functions (e.g., licensed Senior Reactor Operators (SROs), licensed Reactor Operators (ROs), radiation protection technicians, plant equipment 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 been assigned.~~

e. The Shift 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.

5.5.9 Steam Generator (SG) Program (continued)

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair 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.

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 repair 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.

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

Bases

Bases – Tracking Report Revision List

Change ID No.	Section	Page	Reason for change	Change Summary	Rev. of T/R
-	-	-	Incorporate the DCD Chapter 16 changes that are relevant to Part 4 changes.	Incorporate changes as describe in MHI Letter UTR Rev.0 # UAP-HF-09081 dated 03/06/2009 UTR Rev.1 # UAP-HF-09222 dated 04/30/2009 UTR Rev. 3 # UAP-HF-09413 dated 08/03/2009	0
-	-	-	Incorporate the DCD Chapter 16 changes that are relevant to Part 4 changes.	Incorporate changes as describe in MHI Letter UTR Rev.4 # UAP-HF-09454 dated 18/09/2009	1

LCO 3.0.8

LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(~~d~~c)(2)(ii), and, as such, are appropriate for control by the licensee.

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

LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.

LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The RTS is self-tested on a continuous basis from the digital side of all input modules to the digital side of all output modules. Self-testing also encompasses all data communications within a PSMS train, between PSMS trains and between the PSMS and PCMS. The self-testing is described in Reference 6 and 7. The ACTUATION LOGIC TEST is a check of the RTS software memory integrity to ensure there is no change to the internal RTS software that would impact its functional operation or the continuous self-test function. The software memory integrity test is described in Reference 6 and 7. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The complete continuity check from the input device to the output device is performed by the combination of the continuous CHANNEL CHECK, the CHANNEL CALIBRATION for the non digital side of the input module, the continuous self-testing for the digital side, the COT, the ACTUATION LOGIC TEST and the TADOT for the non-digital side of the output module. The CHANNEL CALIBRATION, COT, ACTUATION LOGIC TEST and TADOT, which are manual tests, overlap with the CHANNEL CHECK and self-testing and confirm the functioning of the self-testing.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function and Overpower ΔT Function.

A Note modifies SR 3.3.1.6. 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 Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT.

The RTS is self-tested on a continuous basis from the digital side of all input modules to the digital side of all output modules. Self-testing encompasses all digital Trip Setpoints and trip functions. The self-testing is described in Reference 6 and 7. The COT is a check of the RTS software memory integrity to ensure there is no change to the internal RTS software that would impact its functional operation, including digital Trip Setpoint values or the continuous self-test function. The software memory integrity test is described in Reference 6 and 7.

A COT ensures the entire channel will perform the intended Function. A COT also ensures that the logic processing for interlocks (i.e., P-6 and P-10) is operating correctly. The combination of the COT, CHANNEL CALIBRATION, continuous self-testing and continuous CHANNEL CHECK ensures the comple P-6 and P-10 interlocks are operating correctly.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The complete continuity check from the input device to the output device is performed by the combination of the continuous CHANNEL CHECK, the CHANNEL CALIBRATION for the non digital side of the input module, the continuous self-testing for the digital side, the COT, the ACTUATION LOGIC TEST and the TADOT for the non-digital side of the output module.

SURVEILLANCE REQUIREMENTS (continued)

The CHANNEL CALIBRATION, COT, and TADOT, which are manual tests, overlap with the CHANNEL CHECK and self-testing and confirm the functioning of the self-testing.

The Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTB close for 4 hours this Surveillance must be performed prior to over 4 hours after entry into MODE 3.

SR 3.3.1.8

Performance of the CHANNEL CHECK within 4 hours after reducing power below P-6 and the frequency in accordance with the Surveillance Frequency Control Program ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

BASES

ACTIONS (continued)

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 72 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 5 within the next 30 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. In MODE 5, these Functions are no longer required OPERABLE.

The initial completion time of 72 hours is justified in the PSMS reliability analysis, considering that the remaining operable channels have continuous self-testing. For detail information, refer to the US-APWR Technical Report MUAP-07030 PRA, Attachment 6B.12. The result of the PSMS reliability analysis is evaluated and confirmed in the US-APWR PRA Chapter 19.

L.1

Condition L applies to the Containment Purge Isolation - Actuation Logic and Actuation Output Function and addresses the train orientation of the Engineered Safety Features Actuation System (ESFAS). It also addresses the failure of multiple Containment Purge Isolation - Containment Radiation Monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action K.1.

If an Actuation Logic and Actuation Output train is inoperable, multiple Containment Radiation Monitoring channels are inoperable, or the Required Action and associated Completion Time of Condition K are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

M.1

Condition M applies to the Actuation Logic and Actuation Outputs Function of the MCR Isolation, the Main Control Air monitor Functions, and the Manual Initiation Functions.

If one Actuation Logic and Actuation Outputs train is inoperable, ~~or~~ one Main Control Room Radiation channel is inoperable in one or more Functions, or one Manual Initiation train is inoperable, 7 days are permitted to restore it to OPERABLE status. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this Completion Time is the same as provided in LCO 3.7.10. If the channel/train cannot be restored to OPERABLE status, one train of the effected subsystem(s)~~MCR Isolation train~~ must be placed in the emergency

ACTIONS (continued)

~~radiation protection~~ mode of operation. This accomplishes the actuation instrumentation Function and places the unit in a conservative mode of operation.

Effected subsystems depend on inoperable train, as follows.

- If train A or D is inoperable, MCREFS doesn't satisfy the single failure criterion. Therefore, one train MCREFS is placed on emergency mode. MCRATCS is unaffected, since three required trains remain operable.
- If train B or C is inoperable, MCREFS is unaffected and three required trains of MCRATCS remain operable. Therefore, no action is required.

N.1.1, N.1.2, and N.2

Condition N applies to the failure of two MCR Isolation Actuation Logic and Actuation Outputs trains, two Main Control Room Radiation channels, or two Manual Initiation trains for one or more Functions. The first Required Action is to place the effected subsystem(s)~~two OPERABLE MCR Isolation trains~~ in the emergency mode of operation immediately. For MCREFS this requires one train, since each is 100% capacity. Two trains of MCRATCS are required since each is 50% capacity. This accomplishes the actuation instrumentation Function that may have been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required Actions of LCO 3.7.10 must also be entered for the ~~MCR Isolation~~MCRVS train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed upon train inoperability as discussed in the Bases for LCO 3.7.10.

Alternatively, all trains of the effected subsystem(s)may be placed in the emergency mode. This ensures the MCR Isolation function is performed even in the presence of a single failure. ~~The Required Actions are modified by a Note that excludes this alternative for failure of the Actuation Logic and Actuation Outputs, since a failure of this Function affects normal and emergency modes. This alternative is applicable for failure of the Main Control Room Radiation monitor functions and failure of the Manual Initiation function.~~

Effected subsystems depend on inoperable train, as follows.

- If trains A and D are inoperable, MCREFS is completely inoperable. Therefore, one train of MCREFS is placed on emergency mode and the required action of MCRVS is appled (to restore in 7 days). Or two trains of MCREFS are placed on emergency mode. And one train of MCRATCS is placed on emergency mode, since MCRATCS does not satisfy the single failure criterion.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. The ~~ESFAS~~PSMS is self-tested on a continuous basis from the digital side of all input modules to the digital side of all output modules. Self-testing also encompasses all data communications within a PSMS train, between PSMS trains and between the PSMS and PCMS. The self-testing is described in Reference 6 and Reference 7. The ACTUATION LOGIC TEST is a check of the ESFAS software memory integrity to ensure there is no change to the internal ESFAS software that would impact its functional operation or the continuous self-test function. The software memory integrity test is described in Reference 6 and Reference 7. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The complete continuity check from the input device to the output device is performed by the combination of the continuous CHANNEL CHECK, and the CHANNEL CALIBRATION for the non digital side of the input module, the continuous self-testing for the digital side, the COT, the ACTUATION LOGIC TEST and the ESFAS and SLS TADOT for the non-digital side of the output module. The Channel CALIBRATION, COT, ACTUATION LOGIC TEST and TADOT, which are manual tests, overlap with the CHANNEL CHECK and self-testing and confirm the functioning of the self-testing.

SR 3.3.2.3

SR 3.3.2.3 is the performance of a COT.

The PSMS is self-tested on an automatic basis from the digital side of all input modules to the digital side of all output modules. Self-testing encompasses all Trip Setpoints and trip functions. The self-testing is described in Reference 6 and Reference 7. ESFAS setpoint and bistable functions are implemented within the RPS. Therefore, the COT is a check of the RPS software memory integrity to ensure there is no change to the internal RPS software that would impact its functional operation, including digital Trip Setpoint values or the continuous self-test function. The software memory integrity test is described in Reference 6 and Reference 7.

BASES

SURVEILLANCE REQUIREMENTS (continued)

A COT ensures the entire channel will perform the intended Function.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The complete continuity check from the input device to the output device is performed by the combination of the continuous CHANNEL CHECK, the CHANNEL CALIBRATION for the non digital side of the input module, the continuous self-testing for the digital side, the COT and the TADOT for the non-digital side of the output module. The Channel CALIBRATION, COT and TADOT, which are manual tests, overlap with the CHANNEL CHECK and self-testing and confirm the functioning of the self-testing.

SR 3.3.2.4

SR 3.3.2.4 is the performance of a TADOT for the Actuation Outputs of all ESFAS functions. This function actuates the outputs of the SLS.

Therefore, this test is typically conducted in conjunction with testing the plant process components. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.2.5

SR 3.3.2.5 is the performance of a TADOT for the Loss of Offsite Power, Function. The LOP inputs to the ESFAS are tested up to, and including, the signal status readout on a digital display.

The SR is modified by a Note that excludes verification of setpoints for relays. Relay setpoints require elaborate bench calibration and are verified during CHANNEL CALIBRATION. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

BASES

ACTIONS (Continued)

Required Action C.2 is modified by a Note that indicates C.2 is only required to be performed when the Emergency Feedwater Pit Level is inoperable.

D.1

Condition D applies when the Required Action and associated Completion Time of Condition C is not met. Required Action D.1 requires entering the appropriate Condition referenced in Table 3.3.3-1 for the channel immediately. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met the Required Action of Condition C, and the associated Completion Time has expired, Condition D is entered for that channel and provides for transfer to the appropriate subsequent Condition.

E.1 and E.2

If the Required Action and associated Completion Time of Condition C is not met and Table 3.3.3-1 directs entry into Condition E, the unit must be brought to a MODE where the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

F.1

At this unit, alternate means of monitoring Reactor Vessel Water Level and Containment High Area Radiation have been developed and tested. Also, alternate means of the RCS Hot Leg Temperature (Wide Range) and RCS Cold Leg Temperature (Wide Range) have been developed and tested. Also, alternate means of Steam Generator Water Level (Wide Range) and Emergency Feedwater Flow have been developed and tested. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. If these alternate means are used, the Required Action is not to shut down the unit but rather to follow the directions of Specification 5.6.5, in the Administrative Control section of the TS. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

BASES

SURVEILLANCE REQUIREMENTS (continued)

In SR 3.3.5.3, the values specified for Setpoints will be confirmed following completion of the plant specific setpoint study. These values will be calculated in accordance with the setpoint methodology after selection of plant specific instrumentations.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.5.4

SR 3.3.5.4 is the performance of an ACTUATION LOGIC TEST. The Class 1E GTG start logic within the PSMS is self-tested on a continuous basis from the digital side of all input modules to the digital side of all output modules. Self-testing also encompasses all data communications within a PSMS train, between PSMS trains and between the PSMS and PCMS. The self-testing is described in Reference 2 and 3. The ACTUATION LOGIC TEST is a check of the PSMS software memory integrity to ensure there is no change to the internal PSMS software that would impact its functional operation or the continuous self-test function. The software memory integrity test is described in Reference 2 and 3. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The complete continuity check from the input device to the output device is performed by the combination of the continuous CHANNEL CHECK, the CHANNEL CALIBRATION for the non digital side of the input module, the continuous self-testing for the digital side, the ACTUATION LOGIC TEST, and the ESFAS and SLS TADOT for the non-digital side of the output module. The Channel CALIBRATION, ACTUATION LOGIC TEST and TADOT, which are manual tests, overlap with the CHANNEL CHECK and self-testing and confirm the functioning of the self-testing.

ACTIONS (continued)

C.1 and C.2

If one block valve is inoperable, then it is necessary to restore the block valve to OPERABLE status within the Completion Time of 72 hours. Because at least one SDV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable SDV in Condition B, since the SDVs may not be capable of mitigating an event if the inoperable block valve is not full open. If it cannot be restored within the completion time of 72 hours, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D. Required Action C.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time.

The Required Action C.1 is modified by a Note stating that the Required Actions do not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable ~~PORV~~SDV (which require the block valve power to be removed once it is closed) are adequate to address the condition.

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, then 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 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 and without challenging plant systems.

E.1, E.2, E.3, and E.4

If more than one SDV is inoperable, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If no SDVs are restored within the Completion Time, then 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 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 and without challenging plant systems.

ACTIONS (continued)

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Actions B.1 while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

C.1 and C.2

If ~~a~~the Required Action and the associated Completion Time of Condition A or B is not met, or if the DOSE EQUIVALENT I-131 is > 60 μCi/gm , the reactor must be brought to MODE 3 ~~with RCS average temperature < 500°F~~ within 6 hours and MODE 5 within 36 hours. The allowed Completion Time ~~of 6 hours is~~are reasonable, based on operating experience, to reach the required plant conditions ~~MODE 3 below 500°F~~ from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken ~~a quantitative measure of radionuclides with half lives longer than 15 minutes~~.~~,~~ This Surveillance provides an indication of any increase in the ~~release of~~ noble gas specific activity. ~~from fuel clad defects.~~

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. ~~The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with Tavg at least 500°F.~~ The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.16.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT Xe-133 is not detected, it should be assumed to be present at the minimum detectable activity.

A Note modifies the SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

BASES

SR 3.4.16.2

This Surveillance is performed ~~in MODE 1 only~~ to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when ~~increased releases of iodine from fuel defects are~~iodine spiking is more apt to occur. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the SurveillanceFrequency Control Program. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following ~~fuel failure~~iodine spike initiation; samples at other times would provide inaccurate results.

The Notes modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

REFERENCES

1. 10 CFR 50.34.

2. Standard Review Plan (SRP) Section 15.0.3 “Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors”

3. FSAR ~~Chapter~~Subsection 15.1.5.

4. FSAR Subsection 15.6.3.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.18 RCS Loops – Test Exceptions

BASES

BACKGROUND	The primary purpose of this test exception is to provide an exception to LCO 3.4.4, "RCS Loops - MODES 1 and 2," to permit reactor criticality under no flow conditions during certain PHYSICS TESTS (natural circulation demonstration, station blackout, and loss of offsite power) to be performed while at low THERMAL POWER levels. Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the power plant as specified in GDC 1, "Quality Standards and Records" (Ref. 2). The key objectives of a test program are to provide assurance that the facility has been adequately designed to validate the analytical models used in the design and analysis, to verify the assumptions used to predict plant response, to provide assurance that installation of equipment at the unit has been accomplished in accordance with the design, and to verify that the operating and emergency procedures are adequate. Testing is performed prior to initial criticality, during startup, and following low power operations. The tests will include verifying the ability to establish and maintain natural circulation following a plant trip at low power, performing decay heat removal via natural circulation, and during the natural circulation condition, showing that pressure can be controlled using auxiliary spray and pressurizer heaters powered from the emergency power sources.
APPLICABLE SAFETY ANALYSES	The tests described above require operating the plant without forced convection flow and as such are not bounded by any safety analyses. However, operating experience has demonstrated this exception to be safe under the present applicability. As describe 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.

BASES

LCO	This LCO provides an exemption to the requirements of LCO 3.4.4. The LCO is provided to allow for the performance of PHYSICS TESTS in MODE 2 (after a refueling), where the core cooling requirements are significantly different than after the core has been operating. Without the LCO, plant operations would be held bound to the normal operating LCOs for reactor coolant loops and circulation (MODES 1 and 2), and the appropriate tests could not be performed. In MODE 2, where core power level is considerably lower and the associated PHYSICS TESTS must be performed, operation is allowed under no flow conditions provided THERMAL POWER is ≤ P-7 and the reactor trip setpoints of the OPERABLE power level channels are set ≤ 25% RTP. This ensures, if some problem caused the plant to enter MODE 1 and start increasing plant power, the Reactor Trip System (RTS) would automatically shut it down before power became too high, and thereby prevent violation of fuel design limits. The exemption is allowed even though there are no bounding safety analyses. However, these tests are performed under close supervision during the test program and provide valuable information on the plant's capability to cool down without offsite power available to the reactor coolant pumps.
APPLICABILITY	This LCO is applicable when performing low power PHYSICS TESTS without any forced convection flow. This testing is performed to establish that heat input from nuclear heat does not exceed the natural circulation heat removal capabilities. Therefore, no safety or fuel design limits will be violated as a result of the associated tests.
ACTIONS	A.1 When THERMAL POWER is ≥ the P-7 interlock setpoint 10%, the only acceptable action is to ensure the reactor trip breakers (RTBs) are opened immediately in accordance with Required Action A.1 to prevent operation of the fuel beyond its design limits. Opening the RTBs will shut down the reactor and prevent operation of the fuel outside its design limits.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.18.1

Verification that the power level is < the P-7 interlock setpoint (10%) will ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. The Frequency of once per hour is adequate to ensure that the power level does not exceed the limit. Plant operations are conducted slowly during the performance of PHYSICS TESTS and monitoring the power level once per hour is sufficient to ensure that the power level does not exceed the limit.

SR 3.4.18.2

The power range and intermediate range neutron detectors, P-10, and the P-13 interlock setpoint must be verified to be OPERABLE and adjusted to the proper value. The Low Power Reactor Trips Block, P-7 interlock, is actuated from either the Power Range Neutron Flux, P-10, or the Turbine Inlet Pressure, P-13 interlock. The P-7 interlock is a logic Function with train, not channel identity. A COT is performed 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. The SR 3.3.1.7 Frequency is sufficient for the power range and intermediate range neutron detectors to ensure that the instrumentation is OPERABLE before initiating PHYSICS TESTS, because the RTS is self-tested on a continuous basis from digital side of all input modules to the digital side of all output modules.

SR 3.4.18.3

The Low Power Reactor Trips Block, P-7 interlock, must be verified to be OPERABLE in MODE 1 by LCO 3.3.1, “Reactor Trip System Instrumentation.” The P-7 interlock is actuated from either the Power Range Neutron Flux, P-10, or the Turbine Inlet Pressure, P-13 interlock. The P-7 interlock is a logic Function. An ACTUATION LOGIC TEST is performed to verify OPERABILITY of the P-7 interlock prior to initiation of startup and PHYSICS TESTS. This will ensure that the RTS is properly functioning to provide the required degree of core protection during the performance of the PHYSICS TESTS.

BASES

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50, Appendix A, GDC 1, 1988.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWSP provides a source of borated water to the SI and CS System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Refs. 1 and 2). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "Safety Injection System (SIS) - Operating," B 3.5.3, " Safety Injection System (SIS) - Shutdown," and B 3.6.6, "Containment Spray ~~and Cooling~~ Systems." These analyses are used to assess changes to the RWSP in order to evaluate their effects in relation to the acceptance limits in the analyses.

The RWSP must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWSP, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability.

~~The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a nonlimiting event and the results are very insensitive to boron concentrations.~~ The maximum temperature ~~ensures that the amount of cooling provided from the RWSP during the heatup phase of a feedline break is consistent with safety analysis assumptions~~is an assumption in the steam generator tube rupture analysis; the minimum is an assumption in ~~both~~ the MSLB ~~and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting.~~

For a large break LOCA analysis, the minimum water volume limit of 329,150 gallons and the lower boron concentration limit of 4000 ppm are used to compute the post LOCA boron concentration necessary to assure subcriticality. To secure this minimum water volume in the accident, RWSP needs to store boric acid water ≥ 583,340 gallons during normal operation. This water volume also bounds the ECCS and CSS pump NPSH Requirements. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of 4200 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from direct vessel injection to hot leg injection is to avoid boron precipitation in the core following the accident.

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWSP lower temperature limit of 32°F. If the lower temperature limit is violated, the containment spray further reduces

ACTIONS (continued)

B.1

With the RWSP inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

In this Condition, neither the SIS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWSP is not required. The short time limit of 1 hour to restore the RWSP to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

If the RWSP cannot be returned to OPERABLE status within the associated Completion Time, 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.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWSP borated water temperature should be verified to be within the limits assumed in the accident analyses band. [The Frequency of 24 hours is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ~~ambient~~containment air temperatures are within the operating limits of the RWSP. With ~~ambient~~containment air temperatures within the band, the RWSP temperature should not exceed the limits.

BASES

BACKGROUND (continued)

High Volume Purge System (36 inch purge valves)

The High Volume_Purge System operates to supply outside air into the containment for ventilation and cooling or heating and may also be used to reduce the concentration of noble gases within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. The 36 inch purge valves are normally maintained closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

Low Volume Purge System (8 inch purge valves)

The Low Volume Purge_System operates to:

~~a. Supply outside air into the containment for ventilation and cooling or heating~~

a. Reduce the concentration of noble gases within containment prior to and during personnel access and

b. Equalize internal and external pressures.

Since the valves used in the Low Volume Purge System are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3, and 4.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 1). In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analyses assume that the 36 inch high volume purge valves are closed at event initiation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.6

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B, is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 4).

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

~~SR 3.6.3.6~~SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR ~~Chapter 15~~Subsection 15.6.5.5.

2. FSAR ~~Chapter 6~~Subsection 6.2.4.

3. Standard Review Plan 6.2.4.

4. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND	The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel storage pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Essential Service Water System, and thus to the environment. A typical CCW System is arranged as four independent, 50% capacity cooling loops, and has isolatable nonsafety related components. Each safety related train includes one 50% capacity pump, connection to one of the two surge tanks, a 50% capacity heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus. The surge tanks in the system provide pump trip protective functions to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal, and all nonessential components are isolated. Additional information on the design and operation of the system, along with a list of the components served, is presented in FSAR Chapter 9 (Ref. 1). The principal safety related function of the CCW System is the removal of decay heat from the reactor via the Containment Spray/Residual Heat Removal (CS/RHR) System. This may be during a normal or post accident cooldown and shutdown.
APPLICABLE SAFETY ANALYSES	The design basis of the CCW System is for two CCW trains to remove the post loss of coolant accident (LOCA) heat load from the refueling water storage pit and other components, such as Safety Injection Pumps and CS/RHR Pumps. The Emergency Core Cooling System (ECCS) LOCA and containment OPERABILITY LOCA each model the maximum and minimum performance of the CCW System, respectively. The normal temperature of the CCW is 100°F, and, during unit cooldown to MODE 5 (T_{cold} < 200°F), a maximum temperature of 110°F is assumed. This prevents the refueling water storage pit fluid from increasing in temperature following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System (RCS).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.11.2

This SR verifies that the required annulus emergency exhaust system testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter ~~performance~~, and minimum ~~system~~ flow rate. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.11.3

This SR verifies that the annulus emergency exhaust system starts and operates on an actual or simulated actuation signal. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.7.11.4

The Annulus Emergency Exhaust System produces a negative pressure to prevent leakage from the penetration and safeguard component areas. This SR verifies that the penetration and safeguard component areas can be rapidly drawn down to at a ≤-0.25 inches water gauge relative to atmospheric pressure. This SR verifies the integrity of the penetration and safeguard component areas enclosure. The ability of the penetration and safeguard component areas to draw down and maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of annulus emergency exhaust system. During the accident condition, ~~the~~each annulus emergency exhaust system train is designed to draw down to at a ≤-0.25 inches water gauge relative to atmospheric pressure within 240 seconds after a start signal and maintain a ≤-0.25 inches water gauge relative to atmospheric pressure at a flow rate of 5600 cfm in the associated room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The minimum system flow rate maintains a slight negative pressure in the penetration and safeguard component areas, and provides sufficient air velocity to transport particulate contaminants, assuming only one filter train is operating. The number of filter elements is selected to limit the flow rate through any individual element to about 5600 cfm. This may vary based on filter housing geometry. The maximum limit ensures that the flow through, and pressure drop across, each filter element are not excessive.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.6.4

This Surveillance verifies that the pilot cell temperature is greater than or equal to the minimum established design limit (i.e., 40°F). Pilot cell electrolyte temperature is maintained above this temperature to assure the battery can provide 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 Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The minimum established design limits are determined based on manufacturer's recommendation and controlled by administrative control.

SR 3.8.6.6

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. This test is implemented in accordance with IEEE-450 (Ref. 1).

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.6.6; however, only the modified performance discharge test may be used to satisfy the battery service test requirements of SR 3.8.4.3.

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 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 1) and IEEE-485 (Ref. 5). 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. Manufacturer’s rating of the battery capacity for an acceptance criterion is determined based on manufacturer’s recommendation and controlled by administrative control.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. 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 in accordance with the Surveillance Frequency Control Program. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is reduced in accordance with the Surveillance Frequency Control Program for batteries that retain capacity ≥ 100% of the manufacturer’s ratings. Degradation is indicated, according to IEEE-450 (Ref. 1), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ 10%below the manufacturer’s rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 1). Battery expected life for setting the performance FREQUENCY is determined based on manufacturer’s recommendation and controlled by administrative control.

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.

BASES

REFERENCES	1.	FSAR ~~Chapter 8~~Subsection 8.3.2.
	2.	FSAR Chapter 6.
	3.	FSAR Chapter 15.

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October 21, 2009

U. S. Nuclear Regulatory Commission
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ATTN: David B. Matthews, Director
Division of New Reactor Licensing

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 3 AND 4
DOCKET NUMBERS 52-034 AND 52-035
COMBINED LICENSE APPLICATION UPDATE TRACKING REPORT (PARTS 4, 10, 11)

Dear Sir:

Luminant Generation Company LLC (Luminant) herein submits the second Update Tracking Report (UTR) for Comanche Peak Nuclear Power Plant Units 3 and 4 Combined License Application (COLA) Part 4 and the first UTR for COLA Parts 10 and 11.

Part 4 provides pages revised to reflect changes made in the US-APWR Technical Specifications. Part 10 provides pages revised to reflect additional Inspections, Tests, Analyses and Acceptance Criteria (ITAAC) and license conditions resulting from Requests for Additional Information. Finally, Part 11 provides revised Quality Assurance Program Description pages that reflect Revision 7 to NEI 06-14A, changes in the Luminant corporate structure, and increased consistency with FSAR Chapter 13.

Should you have any questions regarding the UTR, please contact Don Woodlan (254-897-6887, Donald.Woodlan@luminant.com) or me.

There are no commitments in this letter.

I state under penalty of perjury that the foregoing is true and correct.

Executed on October 21, 2009.

Sincerely,

Luminant Generation Company LLC

Rafael Flores

Rafael Flores

Attachment: Combined License Application Update Tracking Report (Parts 4, 10, 11) (on CD)

Electronic Distribution w/attachment

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