DTE Energy
One Energy Plaza, Detroit, MI 48226-1279

DTE Energy

10 CFR 52.79

December 17, 2013
NRC3-13-0034

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

References:
1) Fermi 3
Docket No. 52-033
2) Letter from Jerald G. Head (General Electric-Hitachi Nuclear Energy) to USNRC, "ESBWR Standard Plant Design Certification Application Design Control Document, Revision 10, Tier 1 and Tier 2," MFN-13-100, dated December 11, 2013
3) Letter from Peter W. Smith (DTE Electric Company) to USNRC, "DTE Electric Company Supplemental Response to NRC Request for Additional Information Letter No. 80," NRC3-13-0037, dated December 13, 2013

Subject: DTE Electric Company Submittal of Fermi 3 COLA Markups for Implementation of ESBWR DCD, Revision 10

On December 11, 2013, General Electric-Hitachi (GEH) submitted Revision 10 of the Economic Simplified Boiling Water Reactor (ESBWR) Design Control Document (DCD) to the NRC in Reference 2. During a public teleconference held on September 5, 2013, DTE Electric committed to provide the NRC with markups of the Fermi 3 COLA incorporating the ESBWR DCD, Revision 10 changes within two weeks of the DCD revision being submitted by GEH. This letter transmits those changes.

Markups of the Fermi 3 COLA, Part 2 (FSAR), Part 4 (Technical Specifications and Bases) and Part 10 (License Conditions and ITAAC) are provided in the attachment to this letter. The proposed COLA changes will be incorporated into the next submittal of the Fermi 3 COLA. A brief description of the changes is provided below:

- FSAR Subsection 1.1.1.7, "Incorporation by Reference" is changed to incorporate the ESBWR DCD Revision number.

- Fermi 3 COLA changes necessary to implement ESBWR DCD, Revision 10 as a result of responses to NRC Bulletin 2012-01 were provided in Reference 3.

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- Editorial and formatting changes were made to the Technical Specifications and Bases (Part 4) to conform with the revised ESBWR DCD. Please note that these changes are incorporated into Draft Revision 5 of the Fermi 3 Technical Specifications and Bases, which has been prepared for the upcoming Fermi 3 COLA submittal. The incorporated changes are clearly marked and contain notes explaining the changes.

- Fermi 3 FSAR Subsection 3.9.2.4, "Initial Startup Flow-Induced Vibration Testing of Reactor Internals" was revised and new proposed license conditions were added to Part 10 of the Fermi 3 COLA to address issues associated with BWR steam dryer flow-induced vibration.

If you have any questions, or need additional information, please contact me at (313) 235-3341.

I state under penalty of perjury that the foregoing is true and correct. Executed on the 17th day of December 2013.

Sincerely,

Peter W. Smith, Director
Nuclear Development – Licensing and Engineering
DTE Electric Company

Attachment: Markup of the Fermi 3 COLA

cc: Adrian Muniz, NRC Fermi 3 Project Manager
Tekia Govan, NRC Fermi 3 Project Manager
John Klos, NRC Fermi 3 Project Manager
Bruce Olson, NRC Fermi 3 Environmental Project Manager (w/o attachment)
Fermi 2 Resident Inspector (w/o attachment)
NRC Region III Regional Administrator (w/o attachment)
NRC Region II Regional Administrator (w/o attachment)
Supervisor, Electric Operators, Michigan Public Service Commission (w/o attachment)
Michigan Department of Natural Resources and Environment
Radiological Protection Section (w/o attachment)
Regina A. Borsh, Dominion Energy, Inc.
Barry C. Bryant, Dominion Energy, Inc.

Attachment to
NRC3-13-0034
(following 42 pages)

Markup of the Fermi 3 COLA

The following markup represents how DTE Electric intends to reflect this response in the next submittal of the Fermi 3 COLA. However, the same COLA content may be impacted by responses to other COLA RAIs, other COLA changes, plant design changes, editorial or typographical corrections, etc. As a result, the final COLA content that appears in a future submittal may be different than presented here.

Fermi 3 COLA

Part 2 (FSAR) Markups

1.1.1.7 Incorporation by Reference

10 CFR 52.79 states in part that, "The final safety analysis report need not contain information or analyses submitted to the Commission in connection with the design certification, provided, however, that the final safety analysis report must either include or incorporate by reference the standard design certification final safety analysis report and must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the site characteristics fall within the site parameters specified in the design certification." Therefore, because this COLA references the ESBWR DC application, this FSAR incorporates the ESBWR DCD by reference, with the departures presented in COLA Part 7, and with supplemental information, as appropriate (see Subsection 1.1.1.10). References in this FSAR to the DCD should be understood to mean the ESBWR DCD, Tier 2, submitted by GE-Hitachi Nuclear Energy Americas LLC (GEH), as Revision ~~9~~. 10

1.1.1.8 Departures from the Standard Design Certification (or Application)

A departure is a plant-specific "deviation" from design information in a standard DC rule or, consistent with Section C.III.6 of RG 1.206, from design information in a DC application.

10 CFR 52 clarifies that Tier 2 information in a standard DC rule does not include conceptual design information (CDI) and per Section C.III.6 of RG 1.206, Tier 2 information in a standard DC application does not include CDI. Therefore, replacement or revision of CDI does not constitute a departure. Additionally, information addressing combined license (COL) information/holder items and supplemental information (see Subsection 1.1.1.10) that does not change the intent or meaning of the ESBWR DCD text is not considered a departure from the ESBWR DCD.

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1. Any information which, if lost, misused, modified, or accessed without authorization, can reasonably be foreseen as causing harm to the public interest, the commercial or financial interest of the entity or individual to whom the information pertains, the conduct of NRC and Federal programs, or the personal privacy of individuals. SUNSI has been organized into the following seven groups:
   - Allegation information
   - Investigation information
   - Security-related information
   - Proprietary information
   - Privacy Act information
   - Federal, State, Foreign Government, and international agency information
   - Sensitive internal information

3.9 Mechanical Systems and Components

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.9.2.4 Initial Startup Flow-Induced Vibration Testing of Reactor Internals

Replace the last paragraph with the following.

EF3 COL 3.9.9-1-A

~~The vibration assessment program, as specified in RG 1.20, is provided in DCD Appendix 3L and the following referenced GEH Reports.~~

- ~~NEDE 33259P, "Reactor Internals Flow Induced Vibration Program"~~
- ~~NEDE 33312P, "Steam Dryer – Acoustic Load Definition"~~
- ~~NEDE 33313P, "Steam Dryer – Structural Evaluation"~~
- ~~NEDC 33408P, "ESBWR Steam Dryer – Plant Based Load Evaluation Methodology"~~
- ~~NEDC 33408P, Supplement 1, "ESBWR Steam Dryer – Plant Based Load Evaluation Methodology Supplement 1"~~

Replace with Insert

~~The classification of the Fermi 3 reactor internals in accordance with RG 1.20 is dependent on ESBWR status, i.e. if Fermi 3 is the initial ESBWR to perform testing of the reactor internals, or if testing is performed at another reactor prior to Fermi 3 testing. There are two different scenarios:~~

1. ~~A valid prototype for the Fermi 3 reactor internals does not exist. Under this scenario, Fermi 3 reactor internals is classified as a prototype per Regulatory Guide 1.20.~~
2. ~~A valid prototype for Fermi 3 reactor internals does exist. If the prototype testing is performed outside the United States, the guidance in Regulatory Guide 1.20, Revision 3, Regulatory Position 1.2 would need to be satisfied in order for this reactor to be considered a "valid prototype". Assuming that Fermi 3 reactor internals are substantially similar to the valid prototype and that the valid prototype does not experience inservice problems that result in component or operational modifications, Fermi 3 reactor internals will be classified as non-prototype category I. If any changes to classification for Fermi 3 reactor internals are later determined to be necessary, the classification change will be~~

~~addressed at the time the change is proposed with proper evaluation/justification and documented in a revision to the FSAR.~~

~~**[START COM-FSAR-3.9-001]** The comprehensive vibration assessment program will be developed and implemented as described in DCD Appendix 3L with no departures. The vibration measurement and inspection programs will comply with the guidance specified in RG 1.20, Revision 3, consistent with the Fermi 3 reactor internals classification. A summary of the vibration analysis program and description of the vibration measurement (including measurement locations and analysis predictions) and inspection phases of the comprehensive vibration inspection program will be submitted to the NRC six months prior to implementation. **[END COM-FSAR-3.9-001]**~~

~~**[START COM-FSAR-3.9-006]** The preliminary and final reports (as necessary), which together summarize the results of the vibration analysis, measurement and inspection programs will be submitted to the NRC within 60 and 180 days, respectively, following the completion of the programs. **[END COM-FSAR-3.9-006]**~~

3.9.3.1 Loading Combinations, Design Transients and Stress Limits

Replace the last sentence with the following.

STD COL 3.9.9-2-A **[START COM 3.9-002]** The piping stress reports identified in this DCD section will be completed within six months of completion of DCD ITAAC Table 3.1-1. **[END COM 3.9-002]** **[START COM 3.9-004]** The FSAR will be revised as necessary in a subsequent update to address the results of this analysis. **[END COM 3.9-004]**

3.9.3.7.1(3)e Snubber Preservice and Inservice Examination and Testing

Preservice Examination and Testing

Add the following at the end of this section.

STD COL 3.9.9-4-A A preservice thermal movement examination is also performed; during initial system heatup and cooldown, for systems whose design operating

Insert:

For reactor internals other than the steam dryer, the vibration assessment program, as specified in Regulatory Guide (RG) 1.20, is provided in DCD Appendix 3L and the following referenced GEH Report:

- NEDE-33259P-A, "Reactor Internals Flow Induced Vibration Program"

The classification of the Fermi 3 reactor internals in accordance with RG 1.20 is dependent on ESBWR status, i.e., if Fermi 3 is the initial ESBWR to perform testing of the reactor internals, or if testing is performed at another reactor prior to Fermi 3 testing. There are two different scenarios:

1. A valid prototype for the Fermi 3 reactor internals does not exist. Under this scenario, Fermi 3 reactor internals classification is a prototype per RG 1.20.

2. A valid prototype for Fermi 3 reactor internals does exist. If the prototype testing is performed outside the United States, the guidance in RG 1.20, Revision 3, Regulatory Position 1.2, would need to be satisfied in order for this reactor to be considered a "valid prototype." Assuming that Fermi 3 reactor internals are substantially similar to the valid prototype and that the valid prototype does not experience inservice problems that result in component or operational modifications, Fermi 3 reactor internals will be classified as non-prototype category I. If a change to classification for Fermi 3 reactor internals is later determined to be necessary, the classification change will be addressed at the time the change is proposed with proper evaluation/justification and documented in a revision to the FSAR.

Specific to the steam dryer, the comprehensive vibration assessment program, as specified in RG 1.20, is provided in DCD Appendix 3L and the following referenced GEH Reports:

- NEDE-33312P, "ESBWR Steam Dryer Acoustic Load Definition"
- NEDE-33313P, "ESBWR Steam Dryer Structural Evaluation"
- NEDE-33408P, "ESBWR Steam Dryer – Plant Based Load Evaluation Methodology, PBLE01 Model Description"

The steam dryer is classified as a prototype according to RG 1.20, Revision 3. The following describes the approach for the steam dryer Comprehensive Vibration Assessment Program elements, consistent with Regulatory Guide 1.20:

1. The ESBWR steam dryer Comprehensive Vibration Assessment Program is described in DCD Section 3.9, DCD Appendix 3L, and NEDE-33313P, Section 10.0, which includes a description for preparing and submitting to the NRC a Steam Dryer Monitoring Plan no later than 90 days before startup.
2. As described in NEDE-33313P, Section 10.2(b), an example of a steam dryer predicted analysis that concludes the steam dryer will not exceed stress limits with applicable bias and uncertainties and the minimum alternating stress ratio of 2.0 is provided in NEDE-33408P. Because there currently is no ESBWR as-designed steam dryer, the example of an as-designed steam dryer that has been subject to the predicted analysis process and successful startup testing described in NEDE-33408P serves as the design analysis report for the steam dryer and provides sufficient information for licensing. The post-

licensing commitments in ITAAC and license conditions confirm the acceptability of the ESBWR steam dryer design.

3. The startup program and associated license conditions that include appropriate notification points during power ascension, providing data to the NRC at certain hold points and at full power, and providing to the NRC a full stress analysis report and evaluation within 90 days of reaching the full power level, are established in accordance with NEDE-33313P, Section 10.2(c).
4. Periodic steam dryer inspection during refueling outages is as described in NEDE-33313P, Section 10.2(d), and associated license conditions.

[START COM-FSAR-3.9-001] For reactor internals other than the steam dryer, the comprehensive vibration assessment program will be developed and implemented as described in DCD Appendix 3L with no departures. The vibration measurement and inspection programs will comply with the guidance specified in RG 1.20, Revision 3, consistent with the Fermi 3 reactor internals classification. A summary of the vibration analysis program and description of the vibration measurement (including measurement locations and analysis predictions) and inspection phases of the comprehensive vibration inspection program will be submitted to the NRC six months prior to implementation. For the steam dryer, a Steam Dryer Monitoring Plan will be submitted to the NRC no later than 90 days before startup. **[END COM-FSAR-3.9-001]**

[START COM-FSAR-3.9-006] For reactor internals other than the steam dryer, the preliminary and final reports (as necessary), which together summarize the results of the vibration analysis, measurement and inspection programs will be submitted to the NRC within 60 and 180 days, respectively, following the completion of the programs. For the steam dryer, an analysis of the steam dryer will be submitted to the NRC within 90 days of reaching the full power level. **[END COM-FSAR-3.9-006]**

Fermi 3 COLA

Part 4 (Technical Specifications and Bases) Markups

Table 3.3.7.1-1 (page 1 of 1)
Control Room Habitability Area Heating, Ventilation, and Air Conditioning Subsystem (CRHAVS) Instrumentation

FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION B.1	SURVEILLANCE REQUIREMENTS
1. Control Room Air Intake Radiation – High-High	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4
2. Extended Loss of AC Power	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4
3. Emergency Filter Unit (EFU) Discharge Flow – Low (primary train)	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4
4. EFU Outlet Radiation – High-High (primary train)	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4

Formatting change made in DCD, Revision 10

3.3 INSTRUMENTATION

3.3.7.2 Control Room Habitability Area (CRHA) Heating, Ventilation, and Air Conditioning (HVAC) Subsystem (CRHAVS) Actuation

LCO 3.3.7.2 Three CRHAVS actuation divisions associated with the DC and Uninterruptible AC Electrical Power Distribution Divisions required by LCO 3.8.6, "Distribution Systems - Operating," and LCO 3.8.7, "Distribution Systems – Shutdown," shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required actuation division inoperable.	A.1 Restore required division to OPERABLE status.	12 hours
B. Required Action and associated Completion Time of Condition A not met. OR CRHAVS actuation capability not maintained.	B.1.1 Isolate CRHA boundary. AND B.1.2 Place OPERABLE CRHAVS train in isolation mode. AND B.1.3 Declare remaining CRHAVS train inoperable. OR B.2 Declare affected actuation device(s) inoperable.	Immediately Immediately Immediately Immediately

Formatting change made in DCD, Revision 10

Formatting change made in DCD, Revision 10

BASES

BACKGROUND (continued)

monitoring system for actual plant conditions and is then compared to the predicted value for the cycle exposure.

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in many of the safety analyses in Chapter 15 (Ref. 2). In particular, SDM and reactivity transients, such as control rod withdrawal error events are very sensitive to accurate prediction of core reactivity. These analyses rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity anomaly provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted k_{eff} for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict k_{eff} may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Thereafter, any significant deviations in the measured k_{eff} from the predicted k_{eff} that develop during fuel depletion may be an indication that the assumptions of the design basis transient and accident analyses are no longer valid, or that an unexpected change in core conditions has occurred.

Editorial change made in DCD, Revision 10

Reactivity Anomalies satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the design basis transient and accident analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the difference between the monitored core k_{eff} and the predicted core k_{eff} of ± 1% $\Delta k/k$ has been established based on engineering judgment. A > 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

BASES

LCO (continued)

Although not all control rods are required to be OPERABLE to satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the design basis transient and accident analyses.

APPLICABILITY

In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3, 4, and 5, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod OPERABILITY during these conditions. Control rod requirements in MODE 6 are located in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."

ACTIONS

The ACTIONS Table is modified by two Notes. The first Note allows separate Condition entry for each control rod. This is acceptable since the Required Actions for each Condition provides appropriate compensatory actions for each inoperable control rod. Complying with the Required Actions may allow for continued operation, and subsequent inoperable control rods governed by subsequent Condition entry and application of associated Required Actions. The second Note requires entry into applicable Conditions and Required Actions of LCO 3.7.6, "Selected Control Rod Run-In (SCRRI) and Select Rod Insert (SRI) Functions," when inoperable control rods result in inoperability of the SRI function. This Note is necessary to ensure that the ACTIONS for an inoperable SRI are taken if the control rod inoperability affects the OPERABILITY of the SRI function. Otherwise, pursuant to LCO 3.0.6, these ACTIONS would not be entered even when the LCO 3.7.6 is not met. Therefore, Note 2 is added to require the proper actions are taken.

LMA added in DCD, Revision 10

STD COL 16.0-1-A 3.1.3-1

<u>A.1, A.2, and A.3</u>

Editorial change made in DCD, Revision 10 to be consistent with TS 3.1.3

A control rod is stuck if it will not insert by either FMCRD motor torque or hydraulic scram pressure. A control rod is not made inoperable by a failure of the FMCRD motor if the rod is capable of hydraulic scram. With a fully inserted control rod stuck, no actions are required as long as the control rod remains fully inserted. The Required Actions are modified by a Note that allows a stuck control rod to be bypassed in the Rod Control and Information System (RC&IS) to allow continued operation.

BASES

ACTIONS (continued)

SR 3.3.2.1.9 provides additional requirements when control rods are bypassed in the RC&IS to ensure compliance with the RWE analysis.

STD COL 16.0-1-A 3.1.3-1

The associated control rod drive must be disarmed and isolated within 2 hours. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable amount of time to perform the Required Action in an orderly manner.

The motor drive may be disarmed by bypassing the rod in the RC&IS or disconnecting its power supply. Isolating the control rod from scram prevents damage to the CRD and surrounding fuel assemblies should a scram occur. The control rod can be isolated from scram by isolating it from its associated HCU. Two CRDs sharing an HCU can be individually isolated from scram.

LMA added in DCD, Revision 10

STD COL 16.0-1-A 3.1.3-1

Monitoring of the insertion capability of withdrawn control rods must be performed within 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RC&IS. SR 3.1.3.2 and SR 3.1.3.3 perform periodic tests of the control rod insertion capability of withdrawn control rods. Testing within 24 hours ensures a generic problem does not exist. This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." The Required Action A.2 Completion Time only begins upon discovery of Condition A concurrent with THERMAL POWER greater than the actual LPSP of the RC&IS, since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RC&IS (LCO 3.3.2.1, "Control Rod Block Instrumentation") when below the actual LPSP. The allowed Completion Time of 24 hours from discovery of Condition A, concurrent with THERMAL POWER greater than the LPSP of the RC&IS, provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests.

BASES

ACTIONS (continued)

Formatting change made in Fermi 3 TS Bases to conform with DCD formatting

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a design basis transient or accident require a shutdown, to preserve the single failure criterion, an additional control rod would have to be assumed to fail to insert when required. Therefore, the original SDM demonstration may not be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod withdrawn and the highest worth control rod or control rod pair assumed to be fully withdrawn.

LMA added in DCD, Revision 10

STD COL 16.0-1-A 3.1.3-1

Editorial change made in DCD, Revision 10 to be consistent with TS 3.1.3

The allowed Completion Time of 72 hours to verify SDM is adequate considering that with a single control rod stuck in the withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 5 is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram. Even with the postulated additional single failure of an adjacent control rod to insert, sufficient reactivity control remains to reach and maintain MODE 3 or 4 conditions. In addition, Required Action A.2 performs a movement test on each remaining withdrawn control rod to ensure that no additional control rods are stuck. Therefore, the 72 hour Completion Time to perform the SDM verification in Required Action A.3 is acceptable.

B.1

With two or more withdrawn control rods stuck, the plant must be brought to MODE 3 within 12 hours. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

C.1 and C.2

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Rod Scram Accumulators

BASES

BACKGROUND	The control rod scram accumulators are part of the Control Rod Drive (CRD) System and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a single or pair of control rods associated with a specific hydraulic control unit (HCU) at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free-floating piston. The piston separates the water used to scram the control rods from the nitrogen, which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times of LCO 3.1.4, "Control Rod Scram Times."
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 1, 2, 3, and 4. The design basis transient and accident analyses assume that all of the control rods scram at a specified insertion rate. OPERABILITY of each individual control rod scram accumulator, along with LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.4, ensures that the scram reactivity assumed in the design basis transient and accident analyses can be met. The existence of an inoperable accumulator may invalidate prior scram time measurements for the associated control rods. The scram function of the CRD System, and, therefore, the OPERABILITY of the accumulators, protects the Fuel Cladding Integrity Safety Limit (see Bases for LCO 3.2.2 "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded (see Bases for LCO 3.1.4). Also, the scram function at low reactor vessel pressure (i.e., startup conditions) provides protection against violating fuel design limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). Editorial change made in DCD, Revision 10 Control Rod Scram Accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

Editorial change made in DCD, Revision 10

SURVEILLANCE REQUIREMENTS (continued)

The 24 month Frequency is necessary because of the need to perform this Surveillance during a plant outage. The 24 month Frequency is acceptable because of the low probability that the piping will be blocked due to precipitation of the boron from solution. The saturation temperature of the solution is less than 15.5°C (60°F) (Ref. 4) and requirements in SR 3.1.7.2 conservatively ensure that the SPBS remains above saturation temperature. Additionally, the SLC mixing pump and sample connection may be used to verify flow through the outlet of the accumulator.

SR 3.1.7.10

Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. Isotopic tests on the granular sodium pentaborate to verify the actual B-10 enrichment must be performed prior to addition to the SLC accumulator to ensure that the proper B-10 atom percent is being used.

REFERENCES

1. 10 CFR 50.62.
2. Section 6.3.3.
3. Section 15.4.4.
4. Section 9.3.5.
5. Section 7.8.1.1.
6. Section 3.9.6.1.

Formatting change made in Fermi 3 TS Bases to conform with DCD formatting

BASES

APPLICABLE SAFETY ANALYSES	RPS Manual Actuation does not satisfy any criteria of 10 CFR 50.36(c)(2)(ii), but is retained for the overall redundancy and diversity of the RPS as required by the NRC-approved licensing basis.
LCO	Two manual actuation channels and two Reactor Mode Switch - Shutdown actuation channels as specified in Table 3.3.1.3-1 are required to be OPERABLE to retain the overall redundancy and diversity of the RPS.
APPLICABILITY	The manual actuation Functions are required to be OPERABLE whenever the RPS automatic instrumentation is required to be OPERABLE in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"). RPS is required to be OPERABLE in MODES 1 and 2, and MODE 6 with any control rod withdrawn from a core cell containing one or more fuel assemblies. During normal operation in MODES 3, 4, and 5, all control rods are fully inserted and the Reactor Mode Switch - Shutdown Position control rod withdrawal block (LCO 3.3.2.1, "Control Rod Block Instrumentation") does not allow any control rod to be withdrawn. In MODE 6, control rods withdrawn from a core cell containing no fuel assemblies do not affect the reactivity of the core and therefore are not required to have the capability to scram. Provided all control rods otherwise remain inserted, the RPS function is not required. In this condition the required SDM (LCO 3.1.1,"SHUTDOWN MARGIN") and refuel position one-rod-out/rod-pair-out interlock (LCO 3.9.2, "Refuel Position One-Rod/Rod-Pair-Out Interlock") ensures no event requiring RPS will occur. Under these conditions, the RPS function is not required to be OPERABLE.
ACTIONS	A Note has been provided to modify the ACTIONS related to RPS manual actuation Functions. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the condition. However, the Required Actions for inoperable RPS manual actuation Functions provide appropriate compensatory measures for separate inoperable Functions. As such, a Note has been provided which allows separate Condition entry for each inoperable RPS manual actuation Function.

Formatting change made in Fermi 3 TS Bases to conform with DCD formatting

BASES

ACTIONS (continued)

Formatting change made in Fermi 3 TS Bases to conform with DCD formatting

A.1

If one manual actuation channel is inoperable the capability to shut down the unit with the associated Function is lost. However, manual shutdown capability is retained by the OPERABLE Function. The 12-hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 12-hour Completion Time is acceptable based on engineering judgment considering the availability of the automatic functions and alternative manual trip methods and the low probability of an event requiring manual reactor scram during this interval. The four RPS automatic divisions also have manual trip capability provided by four divisional trip switches that are located in positions easily accessible for optional use by the plant operator.

Alternatively, if it is not desired to place the inoperable channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in a scram), Condition C or D, as appropriate, must be entered and its Required Action taken.

B.1

With one channel of the manual scram Function inoperable and one channel of the Reactor Mode Switch -Shutdown position Function inoperable, the affected channels must be verified in trip immediately. In this Condition, both required manual actuation Functions are inoperable.

Alternatively, if it is not desired to place the inoperable channels in trip (e.g., as in the case where placing the inoperable channels in trip would result in a scram), Condition C or D, as appropriate, must be entered and its Required Action taken.

C.1

With both manual actuation channels inoperable in one or both Functions in MODE 1 or 2 or if any Required Action and associated Completion Time of Condition A or B is not met in MODE 1 or 2, 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 MODE 3 within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.

BASES

ACTIONS (continued)

D.1

With both manual actuation channels inoperable in one or both Functions in MODE 6 or if any Required Action and associated Completion Time of Condition A or B is not met in MODE 6, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Action must continue until all such control rods are fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and, therefore, do not have to be inserted.

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3.1

A CHANNEL FUNCTIONAL TEST is performed on each RPS Manual Scram Function channel to ensure that each channel will perform the intended Function. The Frequency of 7 days is based on the reliability of the RPS actuation logic and controls.

SR 3.3.1.3.2

A CHANNEL FUNCTIONAL TEST is performed on the Reactor Mode Switch - Shutdown Position Function to ensure that the Reactor Mode Switch will perform the intended Function. The Frequency of 24 months is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24-month Frequency.

REFERENCES

None.

Formatting change made in DCD, Revision 10

B 3.3 INSTRUMENTATION

B 3.3.1.5 Neutron Monitoring System (NMS) Automatic Actuation

BASES

BACKGROUND

The NMS Instrumentation provides input to the Reactor Protection System (RPS) when sufficient instrumentation channels indicate a trip condition. The RPS is designed to initiate a reactor scram when one or more monitored parameters exceed their specified limit, to preserve the integrity of the fuel cladding and the Reactor Coolant System (RCS), and minimize the energy that must be absorbed following a loss of coolant accident (LOCA).

A detailed description of the NMS instrumentation and NMS actuation logic is provided in the Bases for LCO 3.3.1.4, "Neutron Monitoring System (NMS) Instrumentation."

This Specification addresses OPERABILITY of the NMS automatic actuation divisions that include the Startup Range Neutron Monitor (SRNM) Trip Logic Units, the Average Power Range Monitor (APRM) Trip Logic Units, which house the Oscillation Power Range Monitor (OPRM) logic, and the associated output to RPS (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"). LCO 3.3.1.4, covers SRNM and APRM (OPRM) channel inputs to the NMS Digital Trip Modules.

Formatting change made in DCD, Revision 10

APPLICABLE SAFETY ANALYSES

The actions of the NMS in conjunction with RPS are assumed in the safety analyses of Reference 1. The NMS provides a trip signal to RPS when monitored parameter values exceed the trip setpoints to preserve the integrity of the fuel cladding, preserve the integrity of the reactor coolant pressure boundary, and preserve the integrity of the containment by minimizing the energy that must be absorbed following a LOCA.

NMS Automatic Actuation satisfies the requirements of Selection Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Three SRNM automatic actuation divisions and three APRM/OPRM automatic actuation divisions are required to be OPERABLE to ensure no single automatic actuation division failure will preclude a scram to occur on a valid signal. The three required divisions are those divisions associated with the DC and Uninterruptible AC Electrical Power Distribution Divisions required by LCO 3.8.6, "Distribution Systems -

BASES

ACTIONS (continued)

E.1 and F.1

If the affected actuation division is not restored to OPERABLE status, or is not in trip, within the allowed Completion Time, or if NMS actuation capability is not maintained, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. The Completion Times are reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

Formatting change made in DCD, Revision 10

SR 3.3.1.5.1

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the NMS automatic actuation divisions. The testing in LCO 3.3.1.1, 3.3.1.2, LCO 3.3.1.4, and the functional testing of control rods in LCO 3.1.3, overlaps this Surveillance to provide complete testing of the assumed safety function.

The 24-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24-month Frequency.

SR 3.3.1.5.2

This SR ensures that the individual required division response times are less than or equal to the maximum values assumed in the accident analysis. The RPS RESPONSE TIME acceptance criteria are included in Reference 3.

STD COL 16.0-1-A 3.3.1.5-2

RPS RESPONSE TIME may be verified by actual response time measurements in any series of sequential, overlapping, or total channel measurements. This test encompasses the NMS automatic actuation divisions that include the SRNM Trip Logic Units, the APRM Trip Logic Units, which house the OPRM logic, and the associated output to RPS. This test overlaps the testing required by SR 3.3.1.4.8 to ensure complete testing of instrument channels and actuation circuitry.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.6.6 is required in MODES 3, 4, and 5. The Frequency for CHANNEL FUNCTIONAL TESTS has been extended from 7 days to 31 days because core reactivity changes do not normally take place in MODES 3, 4, and 5. The 31-day Frequency is based on operating experience and on other Surveillances (such as CHANNEL CHECK) that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

SR 3.3.1.6.7

Performance of a CHANNEL CALIBRATION verifies the performance of the SRNM detectors and associated circuitry. The 24-month Frequency considers the unit conditions required to perform the test, the ease of performing the test, the likelihood of a change in the system or component status. The neutron detectors may be excluded from the CHANNEL CALIBRATION because they cannot readily be adjusted. The detectors are regenerative fission chambers that are designed to have a relatively constant sensitivity over the range, and with an accuracy specified for a fixed useful life.

REFERENCES None.

Formatting change made in DCD, Revision 10

B 3.3 INSTRUMENTATION

B 3.3.5.2 EMERGENCY CORE COOLING SYSTEM (ECCS) ACTUATION

BASES

BACKGROUND

The purpose of the ECCS actuation logic is to initiate appropriate responses from the ECCS to ensure that fuel is adequately cooled in the event of a design basis event.

The ECCS logic actuates the Automatic Depressurization System (ADS), the Gravity-Driven Cooling System (GDCS), the Isolation Condenser System, and Standby Liquid Control (SLC). The equipment involved with ADS is described in the Bases for LCO 3.5.1, "ADS - Operating." The equipment involved with GDCS is described in the Bases for LCO 3.5.2, "Gravity-Driven Cooling System (GDCS) - Operating." The equipment involved with SLC is described in the Bases for LCO 3.1.7, "Standby Liquid Control (SLC) System."

A detailed description of the ECCS instrumentation and ECCS actuation logic is provided in the Bases for LCO 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation."

This specification addresses OPERABILITY of the ECCS actuation circuitry from the outputs of the Digital Trip Module (DTM) functions through the voter logic unit (VLU) functions, the timers, and the load drivers (LDs) associated with the ADS safety relief valves (SRVs), the ADS depressurization valves (DPVs), the GDCS injection valves, the GDCS equalizing line valves, and the SLC squib-actuated valves. Operability requirements associated with the ECCS instrumentation channels are provided in LCO 3.3.5.1. Operability requirements for actuated components (i.e., squibs and solenoid valves) are addressed in LCO 3.1.7, LCO 3.5.1, and LCO 3.5.2, as appropriate.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The actions of the ECCS are explicitly assumed in the safety analyses of Reference 1 and 2. The ECCS is initiated to preserve the integrity of the fuel cladding by limiting the post-LOCA peak cladding temperature to less than the 10 CFR 50.46 limits.

ECCS Actuation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Editorial change made in DCD, Revision 10

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Editorial change made in DCD, Revision 10

The actions of the ICS are explicitly assumed in the safety analyses of Reference 1. The ICS is initiated to preserve the integrity of the fuel cladding by limiting the post-LOCA peak cladding temperature to less than the 10 CFR 50.46 limits. Actuation of the ICS precludes actuation of safety relief valves and limits the peak RPV pressure to less than the ASME Section III Code limits.

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

The OPERABILITY of the ICS is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.5.3-1. Each Function must have the required number of OPERABLE channels, with their setpoints in accordance with the SCP, where appropriate. The actual setpoint is calibrated consistent with the SCP. Each channel must also respond within its assumed response time.

$NTSP_F$s are specified in the SCP, as required by Specification 5.5.11. The $NTSP_F$s are selected to ensure the actual setpoints are conservative with respect to the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the $NTSP_F$, but conservative with respect to its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is non-conservative with respect to its required Allowable Value.

The individual Functions are required to be OPERABLE in the MODES specified in the Table which may require an ICS actuation to mitigate the consequences of a design basis accident or transient.

Although there are four channels of ICS instrumentation for each function, only three ICS instrumentation channels for each function are required to be OPERABLE. The three required channels are those channels associated with the DC and Uninterruptible AC Electrical Power Distribution Divisions required by LCO 3.8.6, "Distribution Systems - Operating," and LCO 3.8.7, "Distribution Systems - Shutdown." This is acceptable because the single-failure criterion is met with three OPERABLE ICS instrumentation channels, and because each ICS instrumentation division is associated with and receives power from only one of the four electrical divisions.

The specific Applicable Safety Analyses, LCO and Applicability discussions are listed below on a Function-by-Function basis.

B 3.3 INSTRUMENTATION

B 3.3.5.4 Isolation Condenser System (ICS) Actuation

BASES

BACKGROUND

The purpose of the ICS actuation logic is to initiate appropriate actions to ensure ICS operates following a reactor pressure vessel (RPV) isolation after a scram to provide adequate RPV pressure reduction to preclude safety relief valve operation and to conserve RPV water level to avoid automatic depressurization caused by low water level. In addition, in the event of a loss of coolant accident (LOCA), the ICS instrumentation ensures the system operates to provide additional liquid inventory to the RPV upon opening of the condensate return valves. The ICS actuation logic also ensures the ICS is vented to mitigate the accumulation of radiolytic hydrogen and oxygen in order to prevent a detonation.

A detailed description of the ICS actuation instrumentation is provided in the Bases for LCO 3.3.5.3, "Isolation Condenser System (ICS) Instrumentation."

This specification addresses OPERABILITY of the ICS actuation circuitry from the outputs of the Digital Trip Module (DTM) functions through the voter logic unit (VLU) functions, the timers and the load drivers (LDs) associated with the ICS. Operability requirements associated with ICS instrumentation channels are provided in LCO 3.3.5.3. Operability requirements for actuated components are addressed in LCO 3.5.4, "Isolation Condenser System (ICS) - Operating."

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Editorial change made in DCD, Revision 10

The actions of the ICS are explicitly assumed in the safety analyses of Reference 1. The ICS is initiated to preserve the integrity of the fuel cladding by limiting the post-LOCA peak cladding temperature to less than the 10 CFR 50.46 limits. Actuation of the ICS also, precludes actuation of safety relief valves and limits the peak RPV pressure to less than the ASME Section III Code limits.

ICS actuation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Although there are four divisions of ICS actuation, only three ICS actuation divisions for each function are required to be OPERABLE. The three required divisions are those divisions associated with the DC and Uninterruptible AC Electrical Power Distribution Divisions required by LCO 3.8.6, "Distribution Systems - Operating," and LCO 3.8.7, "Distribution Systems - Shutdown." This is acceptable because the

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

In MODES 5 and 6, low RPV water level may indicate a loss of coolant. Should RPV water level decrease too far, the ability to cool the core may be threatened. Closure of the RWCU/SDC isolation valves isolates the system from the RPV, minimizing the potential loss of coolant inventory. The Reactor Vessel Water Level - Low, Level 2 is implicitly credited in the shutdown probabilistic risk assessment (Ref. 3), and therefore satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

Formatting change made in DCD, Revision 10

Reactor Vessel Water Level - Low, Level 2 signals are initiated from four level sensors that sense the difference between the pressure due to a constant column (reference leg) of water and the pressure due to the actual water level (variable leg) in the vessel. Three channels of Reactor Vessel Water Level - Low, Level 2 Function are required to be OPERABLE to ensure no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level - Low, Level 2 Allowable Value was chosen to be the same as the Isolation Condenser System Reactor Vessel Water Level - Low, Level 2 Allowable Value.

This Function isolates the RWCU/SDC lines, Equipment and Floor Drain System lines, Containment Inerting System lines, and the Fuel and Auxiliary Pools Cooling System process lines.

2. Reactor Vessel Water Level - Low, Level 1

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The isolations of valves whose penetration communicate with the containment or the reactor vessel and the isolation of the reactor building boundary isolation dampers limit the release of fission products to help ensure that offsite does limits are not exceeded. The Reactor Vessel Water Level - Low, Level 1 channels are provided as a backup to the Reactor Vessel Water Level - Low, Level 2 channels and is not credited in the safety analysis.

Reactor Vessel Water Level - Low, Level 1 signals are initiated from four level sensors that sense the difference between the pressure due to a constant column (reference leg) of water and the pressure due to the actual water level (variable leg) in the vessel. Three channels of Reactor Vessel Water Level - Low, Level 1 Function are required to be OPERABLE to ensure no single instrument failure can preclude the isolation function.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Editorial change made in DCD, Revision 10

The ability of the CRHAVS to maintain habitability of the CRHA is an explicit assumption for the safety analyses presented in Chapter 6 and Chapter 15, (Refs. 1 and 3, respectively). The isolation mode of the CRHAVS is assumed to operate following a design basis accident (DBA). The radiological dose to control room personnel as a result of various DBAs is summarized in Reference 3. No single active failure will result in a loss of the system design function.

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

The OPERABILITY of the CRHAVS instrumentation is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.7.1-1. Each Function must have the required number of OPERABLE channels, with their setpoints in accordance with the SCP, where appropriate.

$NTSP_F$s are specified in the SCP, as required by Specification 5.5.11. The $NTSP_F$s are conservative with respect to the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the $NTSP_F$, but conservative with respect to its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is non-conservative with respect to its required Allowable Value.

The individual Functions are required to be OPERABLE in MODES 1, 2, 3, and 4 to maintain habitability of the control room following a DBA, since the DBA could lead to a fission product release.

In MODES 5 and 6, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations in these MODES. Therefore, the Functions listed in Table 3.3.7.1-1 are not required to be OPERABLE in MODES 5 or 6, except for other situations under which significant radioactive releases can be postulated, i.e., during operations with a potential for draining the reactor vessel (OPDRVs).

Although there are four channels of CRHAVS instrumentation for each function, only three channels of CRHAVS instrumentation for each function are required to be OPERABLE. The three required channels are those channels associated with the DC and Uninterruptible AC Electrical Power Distribution Divisions required by LCO 3.8.6, "Distribution Systems – Operating," and LCO 3.8.7, "Distribution Systems – Shutdown." This is acceptable because the single-failure criterion is met with three

B 3.3 INSTRUMENTATION

B 3.3.7.2 Control Room Habitability Area (CRHA) Heating, Ventilation, and Air Conditioning (HVAC) Subsystem (CRHAVS) Actuation

BASES

BACKGROUND

The purpose of the CRHAVS actuation logic is to initiate appropriate actions to ensure the CRHAVS and control room habitability area (CRHA) boundary provide a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity. The equipment involved with CRHAVS is described in the Bases for LCO 3.7.2, "Control Room Habitability Area (CRHA) Heating, Ventilation, and Air Conditioning (HVAC) Subsystem (CRHAVS)."

EF3 COL 16.0-1-A 3.3.7.2-1

This specification addresses OPERABILITY of the CRHAVS actuation circuitry from the outputs of the Digital Trip Module (DTM) functions through the voter logic unit (VLU) functions and the load drivers (LDs) associated with the CRHAVS. Operability requirements associated with the CRHAVS instrumentation channels are provided in LCO 3.3.7.1, "Control Room Habitability Area (CRHA) Heating, Ventilation, and Air Conditioning (HVAC) Subsystem (CRHAVS) Instrumentation." Operability requirements for actuated components (i.e., dampers and valves) are addressed in LCO 3.7.2, "Control Room Habitability Area (CRHA) Heating, Ventilation, and Air Conditioning (HVAC) Subsystem (CRHAVS)."

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Editorial change made in DCD, Revision 10

The ability of the CRHAVS to maintain habitability of the CRHA is an explicit assumption for the safety analyses presented in Chapter 6 and Chapter 15, (Refs. 1 and 2, respectively). The isolation mode of the CRHAVS is assumed to operate following a design basis accident (DBA). The radiological dose to control room occupants as a result of various DBAs is summarized in Reference 2. No single active failure will result in a loss of the system design function.

CRHAVS actuation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Editorial change made in DCD, Revision 10

1.a, 2.a Reactor Vessel Level – Low, Level 1

Automatic actuation of ADS (consisting of the SRVs and DPVs) and GDCS injection occurs upon detection of Reactor Vessel Level – Low, Level 1. Reactor Vessel water level is detected by four wide range water level sensors that are different from those used for the SSLC/ESF wide range level sensors. Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result.

The Reactor Vessel Level – Low, Level 1 Function is required to be OPERABLE in MODES 1, 2, 3, and 4, consistent with the assumptions in Reference 1.

1.b, 2.b Drywell Pressure – High (Manual Actuation)

Manual controls are provided for ADS (consisting of the SRVs and DPVs) and GDCS injection initiation upon detection of high drywell pressure sustained for 60 minutes. This control is provided to mitigate small and medium break LOCA scenarios that do not result in GDCS and ADS initiation from low RPV water level. This Function also requires OPERABILITY of DPS indication of the high drywell pressure condition.

The Drywell Pressure – High (Manual Actuation) Function is required to be OPERABLE in MODES 1, 2, 3, and 4 consistent with the assumptions in Reference 1.

3.a. Reactor Vessel Level – Low (Manual Actuation)

Manual controls are provided for initiation of the GDCS equalizing lines upon detection of low reactor vessel water level. Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. This Function also requires OPERABILITY of DPS indication of the low water level condition.

The Reactor Vessel Level – Low (Manual Actuation) Function is required to be OPERABLE in MODES 1, 2, 3, and 4, consistent with the assumptions in Reference 1.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Editorial change made in DCD, Revision 10

4.a Reactor Water Cleanup/Shutdown Cooling System Differential Mass Flow – High

Automatic isolation of RWCU/SDC occurs upon detection of Reactor Water Cleanup/Shutdown Cooling System Differential Mass Flow – High. Isolation of the RWCU System is initiated when RWCU/SDC System Differential Mass Flow – High is sensed to prevent exceeding off-site doses.

The function of the RWCU/SDC isolation valves, in combination with other accident mitigation systems, is to limit fission product release during a postulated Design Bases Accident (DBA).

The Reactor Water Cleanup/Shutdown Cooling System Differential Mass Flow – High Function is required to be OPERABLE in MODES 1, 2, 3, and 4, consistent with the assumptions in Reference 1.

5.a Isolation Condenser/Passive Containment Cooling System Pool Level – Low

Automatic actuation of the IC/PCCS expansion pool-to-equipment pool cross-connect occurs upon detection of Isolation Condenser/Passive Containment Cooling System Pool Level – Low in the associated IC/PCCS inner expansion pool. Actuation of the IC/PCCS expansion pool-to-equipment pool cross-connect ensures a sufficient quantity of water is available for decay heat removal in the event of a design basis accident.

The Isolation Condenser/Passive Containment Cooling System Pool Level – Low Function is required to be OPERABLE in MODES 1, 2, 3, and 4, consistent with the assumptions in Reference 1.

ACTIONS

A Note has been provided to modify the ACTIONS related to the DPS Functions. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the condition. However, the Required Actions for inoperable DPS Functions provide appropriate compensatory measures for separate

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

STD COL 16.0-1-A 3.4.4-1

The PTLR contains P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic testing, and data for the maximum rate of change of reactor coolant temperature. The heatup curve provides limits for both heatup and criticality.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component of most concern in regard to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 2).

Formatting change made in DCD, Revision 10

The actual shift in the Reference Temperature, Nil-Ductility Transition (RT_{NDT}) of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and 10 CFR 50, Appendix H (Ref. 4). The operating P/T limit curves will be adjusted as necessary, based on the evaluation findings and the recommendations of Reference 5.

BASES

ACTIONS (continued)

E.1 and E.2

If the LCO is not met for reasons other than Condition A, B, or C, action must be initiated to provide at least two methods of injecting the minimum specified volume of water into the RPV. In addition, LCO requirements must be met within 72 hours. This Completion Time is based on engineering judgment considering the low probability of an event requiring GDCS injection when in this Condition.

Alternate sources and methods for water injection are identified in the plant's Abnormal and Emergency Operating Procedures. The method used to provide water for core flooding is based on plant conditions.

F.1, F.2.1.and F.2.2

If Required Actions and associated Completion Times are not met, the water inventory available for injection may not be sufficient to respond to a loss of decay heat removal capability, LOCA, or inadvertent vessel draindown. Therefore, actions to suspend operations with a potential for draining the reactor vessel (OPDRVs) must be initiated immediately to minimize the probability of a vessel draindown. Actions must continue until OPDRVs are suspended. In addition, action must be initiated immediately to establish reactor building refueling and pool area HVAC subsystem (REPAVS) and contaminated area HVAC subsystem (CONAVS) area isolation boundary. This can be accomplished by isolating the REPAVS and CONAVS dampers or verifying the automatic isolation capability of the respective exhaust high radiation function. This action is needed to establish appropriate compensatory measures for a potential loss of decay heat removal as a result of an inadvertent draindown event. The Completion Times are based on engineering judgment considering the need for prompt action to mitigate the consequences of a potential loss of decay heat removal capability, LOCA, or inadvertent vessel draindown.

Formatting change made in DCD, Revision 10

B 3.5 Emergency Core Cooling Systems (ECCS)

B 3.5.5 Isolation Condenser System (ICS) - Shutdown

BASES

BACKGROUND	The ICS is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation to provide adequate RPV pressure reduction to preclude safety relief valve operation and provide core cooling while conserving reactor water inventory (Ref. 1). A description of the ICS is provided in the Bases for LCO 3.5.4, "Isolation Condenser System (ICS) - Operating." When the reactor is shutdown, a reduced ICS capability is maintained to provide cooldown capability and to ensure a highly reliable and passive alternative to the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) system for decay heat removal. RWCU/SDC consists of two independent and redundant trains powered from separate electrical divisions that can be powered from either offsite power or the standby diesel generators. However, RWCU/SDC is a nonsafety-related system that cannot be assumed to remain available following an equipment failure or a loss of offsite power. Depending on plant and equipment status, various alternatives to the RWCU/SDC for decay heat removal can be configured in MODES 3, 4 and 5. When the Isolation Condenser/Passive Containment Cooling System (IC/PCCS) pool and the individual ICS pool subcompartments are flooded, use of one or more ICS loops is the preferred backup method for decay heat removal in MODES 3 and 4. Although not effective for decay heat removal in MODE 5, the ICS does provide a highly reliable and passive backup to the RWCU/SDC for decay heat removal in this MODE. If normal decay heat removal capability is lost, the reactor coolant temperature will increase until the ICS provides the required decay heat removal capacity.
APPLICABLE SAFETY ANALYSES	A highly reliable, safety-related, and passive alternative to RWCU/SDC for decay heat removal when shutdown is not required for mitigation of any event or accident evaluated in the safety analyses. However, decay heat removal must be accomplished to prevent core damage. ICS - Shutdown satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

Editorial change made in DCD, Revision 10

BASES

APPLICABLE SAFETY ANALYSES (continued)

Suppression pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that suppression pool water level be maintained ≥ 5.4 meters (17.7 feet) and ≤ 5.5 meters (18.0 feet) above the pool floor. These limits ensure that the initial conditions assumed for the safety analyses for containment are met.

APPLICABILITY

Suppression pool water level must be maintained within specified limits in MODES 1, 2, 3, and 4 when a DBA could cause significant loads on the containment. In MODES 5 and 6, the potential for SRV actuation is eliminated and the probability and consequences of LOCA are reduced because RPV pressure and temperature are lower. Therefore, maintaining suppression pool level within limits is not required to ensure containment integrity when in MODE 5 or 6.

ACTIONS

<u>A.1</u>

If suppression pool water level is not within specified limits, the initial conditions assumed for the safety analyses are not met. Therefore, suppression pool water level must be restored to within specified limits within 2 hours. This Completion Time is expected to be sufficient to restore suppression pool water level.

The 2-hour Completion Time is acceptable because the pressure suppression function still exists as long as the main vents, SRV quenchers, and PCCS vent return lines are covered even if water level is below the minimum level. Additionally, protection against overpressurization may still exist due to the margin in the peak containment pressure analysis even if water level is above the maximum level. This Completion Time also takes into account the low probability of an event during this interval.

Editorial change made in DCD, Revision 10

B 3.7 PLANT SYSTEMS

B 3.7.6 Selected Control Rod Run-In (SCRRI) and Select Rod Insert (SRI) Functions

BASES

BACKGROUND

Selected Control Rod Run-In (SCRRI) function logic is performed when the Rod Control and Information System (RC&IS) performs 2/3 voting on a Selected Control Rod Run-in/Select Rod Insert (SCRRI/SRI) signal from the Diverse Protection System (DPS) (Ref. 1). RC&IS provides for electrical insertion of selected control rods: 1) for mitigation of a loss of feedwater heating event; or 2) for providing needed power reduction after occurrence of a load rejection event or a turbine trip event. The Automated Thermal Limit Monitor (ATLM) provides an additional SCRRI/SRI signal to RC&IS for mitigation of a loss of feedwater heating event.

RC&IS utilizes a dual-redundant architecture of two independent channels for normal monitoring of control rod positions and executing normal control rod movement commands. Under normal conditions, each channel receives separate input signals and both channels perform the same functions. For the Fine Motion Control Rod Drive (FMCRD) emergency insertion functions (scram-follow, FMCRD run-in, and SCRRI), 3-out-of-3 logic is used in the induction motor controller logic with the additional input signal coming from the associated emergency rod insertion panels. An automatic single channel bypass feature (only activated when an emergency insertion function is activated) is also provided to assure high availability for the emergency insertion functions when a single channel failure condition exists.

Failure or malfunction of RC&IS has no impact on the hydraulic scram function of the CRDs. The circuitry for normal insertion and withdrawal of control rods in RC&IS is completely independent of the Reactor Protection System (RPS) circuitry controlling the scram valves. This separation of the RPS scram and RC&IS normal rod control functions prevents failure in the RC&IS circuitry from affecting the scram circuitry.

Select Rod Insert (SRI) function logic in DPS produces the automatic SRI command signal to the scram timing test panel (Ref. 1). Similarly, 2/3 voting is performed by the DPS on the hard-wired turbine trip and load reject signals from the turbine control system to produce an automatic SRI command signal to the scram timing test panel. The scram timing test panel provides for hydraulic scram insertion of selected control rods: 1) for mitigation of a loss of feedwater heating event; or 2) for providing

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BASES

BACKGROUND (continued)

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needed power reduction after occurrence of a load rejection event or a turbine trip event. ATLM provides an additional SCRRI/SRI signal to RC&IS for mitigation of a loss of feedwater heating event.

DPS utilizes a triplicate redundant system to produce the SRI signal to the scram timing test panel, which on a valid SRI initiation signal causes all the hydraulic control unit (HCU) solenoid return line switches for the control rods selected for SRI to open, resulting in a hydraulic scram of those control rods. The scram timing test panel allows specific HCUs associated with the predetermined SRI control rods to be selected on the scram timing test panel video display unit interface.

Failure or malfunction of DPS or the scram timing test panel has no impact on the hydraulic scram function of the CRDs. The circuitry for emergency electrical insertion and SRI hydraulic insertion of control rods in DPS and the scram timing test panel is completely independent of the RPS circuitry controlling the scram valves. This separation of the RPS scram and the DPS and scram timing test panel control rod functions prevents failure in the DPS and scram timing test panel circuitry from affecting the scram circuitry.

APPLICABLE SAFETY ANALYSES

The SCRRI and SRI functions are assumed to function during transient events that could result in a decrease in core coolant temperature or increase in reactor pressure (i.e., loss of feedwater heating, generator load rejection, and turbine trip). Power reduction from the electrical run-in and hydraulic insertion of selected control rods during these events mitigates the decrease in the MCPR during the event.

STD COL 16.0-1-A 3.7.6-1

The SCRRI and SRI functions satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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BASES

ACTIONS (continued)

72 hours. In this condition, continued power operation should not exceed 72 hours. The 72 hour Completion Time for the restoration of the inoperable DC source is consistent with the time allowed for one inoperable DC Electrical Power Distribution bus.

B.1

Condition B represents both DC Sources inoperable on one required division. In this Condition, the affected division of the DC Sources may not have adequate capacity to support the associated division of the DC Electrical Power Distribution system following a transient event or DBA concurrent with a loss of offsite and onsite AC power.

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With both DC Sources inoperable on one required division, the two remaining required divisions of DC and Uninterruptible AC Electrical Power have the capacity to support a safe shutdown and to mitigate an accident condition even if power is lost to the supporting isolation power center buses. However, a single failure could result in the loss of minimum necessary 250 VDC subsystems. Therefore, continued power operation should not exceed 8 hours. The 8 hour Completion Time for the restoration of an inoperable DC source is consistent with the time allowed for an inoperable division of DC Electrical Power Distribution.

C.1 and C.2

When one or more DC Sources on two or more required divisions are inoperable, the remaining DC Sources may not have the capacity to supply power to the divisions of the DC Electrical Power Distribution system for the required duration of 72 hours following a transient event or DBA, concurrent with a loss of offsite and onsite AC power. If the Required Actions for restoration cannot be met within the specified Completion Times, the plant remains vulnerable to a single failure that could impair the capability to reach safe shutdown or to mitigate an accident condition. Therefore, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 5 within

BASES

APPLICABLE SAFETY ANALYSES

The refueling interlocks are explicitly assumed in the safety analysis of the control rod removal error during refueling (Ref. 3). This analysis evaluates the consequences of control rod withdrawal during refueling. A prompt reactivity excursion during refueling could potentially result in fuel failure with subsequent release of radioactive material to the environment.

Criticality and, therefore, subsequent prompt reactivity excursions are prevented during the insertion of fuel, provided all control rods are fully inserted during the fuel insertion. The refueling interlocks accomplish this by preventing loading fuel into the core with any control rod withdrawn, or by preventing withdrawal of a rod from the core during fuel loading.

Refueling Equipment Interlocks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

To prevent criticality during refueling, the refueling interlocks associated with the reactor mode switch in Refuel position ensure that fuel assemblies are not loaded into the core with any control rod withdrawn.

Editorial change made in DCD, Revision 10

To prevent these conditions from developing, the all-rods-in, the refueling machine position, and the refueling machine fuel grapple hoist fuel-loaded (or auxiliary hoist fuel-loaded, if being used) inputs are required to be OPERABLE. These inputs are combined in logic circuits that provide refueling equipment or control rod blocks to prevent operations that could result in criticality during refueling operations.

APPLICABILITY

In MODE 6, a prompt reactivity excursion could cause fuel damage and subsequent release of radioactive material to the environment. The refueling equipment interlocks protect against prompt reactivity excursions during MODE 6. The interlocks are only required to be OPERABLE during in-vessel fuel movement with refueling equipment associated with the interlocks when the reactor mode switch is in the Refuel position.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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The refuel position one-rod/rod-pair-out Interlock satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

To prevent criticality during MODE 6, the refuel position one-rod/rod-pair-out interlock ensures no more than one control rod or one control rod pair with the same HCU may be withdrawn. The refuel position one-rod/rod-pair-out interlock is required to be OPERABLE and the reactor mode switch must be locked in the refuel position to support the OPERABILITY of the interlock.

APPLICABILITY

In MODE 6, with the reactor mode switch in the refuel position, the OPERABLE refuel position one-rod/rod-pair-out interlock provides protection against prompt reactivity excursions.

In MODES 1, 2, 3, 4 and 5, the refuel position one-rod/rod-pair-out interlock is not required to be OPERABLE and is bypassed. In MODES 1 and 2, the Reactor Protection System (RPS) (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," LCO 3.3.1.2, "Reactor Protection System (RPS) Actuation," and LCO 3.3.1.3, "Reactor Protection System (RPS) Manual Actuation") and the control rods (LCO 3.1.3, "Control Rod OPERABILITY") provide mitigation of potential reactivity excursions. In MODES 3, 4 and 5, with the reactor mode switch in the shutdown position, a control rod block (LCO 3.3.2.1, "Control Rod Block Instrumentation") ensures all control rods are inserted, thereby preventing criticality during shutdown conditions.

ACTIONS

A.1 and A.2

With the refuel position one-rod/rod-pair-out interlock inoperable, the refueling interlocks may not be capable of preventing more than one control rod or control rod pair from being withdrawn. This condition may lead to criticality.

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Part 10 (License Conditions and ITAAC) Markups

Insert the Following Proposed License Conditions at the End of Part 10:

3.10 Steam Dryer License Conditions

The licensee will use supporting information in Reports NEDE-33312P and NEDE-33313P for implementing the actions associated with the following license conditions.

Steam Dryer License Conditions:

1.a. A Steam Dryer Monitoring Plan (SDMP) for the steam dryer will be prepared and provided to the NRC no later than 90 days before startup.

1.b. Power Ascension Test (PAT) procedures for the steam dryer testing will be provided to NRC inspectors no later than 10 days before start-up. The PAT procedures will include the following:

- Level 1 and Level 2 acceptance limits for on-dryer strain gages, and on-dryer accelerometers to be used up to 100% power.
- Specific hold points and their duration during 100% power ascension.
- Activities to be accomplished during hold points.
- Plant parameters to be monitored.
- Actions to be taken if acceptance criteria are not satisfied.
- Verification of the completion of commitments and planned actions.

2. An initial hold point during the first plant power ascension will be at no more than 75 percent of full power. At this hold point, the licensee will:

    a. Record data from the on-dryer mounted instrumentation. Evaluate data and compare to acceptance limits in the PAT procedures.

    b. Develop a flow-induced vibration (FIV) load definition, using ESBWR PBLE01 methodology, based on selected on-dryer instruments. Using appropriate methods and the FIV load definition, predict the steam dryer response at this condition.

    c. Compare the predicted steam dryer strains and accelerations against the measured data. If any of the measured sensor data exceed the adjusted predictions, then either modify the limit curves and ensure measured sensor responses do not exceed the adjusted predictions, or evaluate the impact on dryer fatigue life.

    d. Define the steam dryer peak stress projections based on the revised results from step 2(b) with adjustments from step 2(c). Compute the steam dryer maximum stress and minimum stress ratio from the predictive analysis and plot stresses as described in the applicable engineering reports. The adjusted peak stress amplitude is maintained less than 93.7 MPa (13,600 psi).

    e. Update limit curves based on the results from step 2(d). Level 1 and Level 2 limit curves will be generated for all functioning strain gage and accelerometer locations.

f. Trend the recorded data and project sensor responses for the next assessment point and full power to demonstrate margin for continued power ascension.

g. Make available to the NRC the ESBWR steam dryer analysis summary, updated stress analysis results, limit curves, and data projections for higher power levels.

3. Continue power ascension with subsequent hold points at approximately 5 percent power level increments where data will be recorded and evaluated. Data trending and projections will be generated for the next hold point and full power. Data trending analysis during power ascension must assess whether the limits would be violated at higher power levels. Data trending results and revised limit curves will be made available to the NRC at each hold point.

4. Power ascension monitoring shall address expected increases in loading and fatigue damage due to variable plant conditions throughout the life of the dryer.

5. During initial power ascension, if flow-induced resonances are identified and the strains or vibrations increase above the pre-determined criteria, power ascension is stopped. The acceptability of the steam dryer for continued operation is evaluated. The limit curves are then redefined based on the on-dryer data. The limit curve factor is revised. If a Level 1 limit curve is exceeded, power will be reduced to a previous power level where Level 1 was not exceeded and a stress analysis will be performed to develop new limit curves. During initial power ascension, should a Level 2 limit curve be exceeded, or if the trending indicates that a Level 1 limit may be challenged prior to reaching the next hold point, the acceptance limits will be evaluated, and revised if appropriate.

6. Compare the predicted and measured strain or acceleration on the steam dryer at each hold point to confirm the conservatism of the predicted dryer stress field. Adjust the predicted strain and acceleration responses. If any of the measured sensor data exceed the adjusted predictions, then either modify limit curves and ensure measured sensor responses do not exceed the adjusted predictions, or evaluate the impact on dryer fatigue life.

7. At the initial hold point and the hold points at approximately 85 and 95 percent power, power ascension will not proceed for at least 72 hours after making the steam dryer data analysis and results available to the NRC, unless notified by the NRC that power ascension may proceed earlier than 72 hours.

8. During the Power Maneuvering in the Feedwater Temperature Operating Domain testing, pressures, strains, and accelerations will be recorded from the on-dryer mounted instrumentation across the expected range of normal steady state plant operating conditions. An evaluation of the dryer structural response over the range of steady state plant operating conditions will be included in the stress analysis report described in Item 9 below.

9. After full power has been achieved, data at the full power level will be provided to the NRC within 72 hours, and a full stress analysis report and evaluation will be provided to the NRC within 90 days of reaching the full power level. The report shall contain information to demonstrate that the steam dryer will maintain its structural integrity over

its design life considering variations in plant parameters (such as reactor pressure and core flow rate).

10. A periodic steam dryer inspection program will be implemented as follows:

a. During the first two scheduled refueling outages after reaching full power conditions, a visual inspection will be conducted of all accessible areas and susceptible locations of the steam dryer in accordance with accepted industry guidance on steam dryer inspections. The results of these baseline inspections will be provided to the NRC within 60 days following startup after each outage.

b. At the end of the second refueling outage following full power operation, an updated SDMP reflecting a long-term inspection plan based on plant-specific and industry operating experience will be provided to the NRC within 180 days following startup from the second refueling outage.