Dominion Resources Services, Inc.
5000 Dominion Boulevard, Glen Allen, VA 23060
Web Address: www.dom.com

Dominion®

February 10, 2014

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Serial No. NA3-14-001
Docket No. 52-017
COL/BCB

DOMINION VIRGINIA POWER
NORTH ANNA UNIT 3 COMBINED LICENSE APPLICATION
COLA MARKUPS FOR IMPLEMENTATION OF 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. During an ESBWR Design Center Working Group public meeting on December 17, 2013, Dominion committed to provide the NRC with markups of the North Anna Unit 3 (NA3) Combined License Application (COLA) that incorporate the ESBWR DCD, Revision 10 changes. This letter transmits those NA3 COLA markups.

Markups of COLA Part 2, Final Safety Analysis Report (FSAR); Part 4, Technical Specifications, and Part 10, Tier 1/ITAAC/Proposed License Conditions, are provided in the enclosure to this letter. A brief description of the proposed changes is provided below:

- FSAR Section 1.1.1.7, "Incorporation by Reference," is being changed to reflect the latest ESBWR DCD revision number (i.e., Rev. 10).
- FSAR Section 3.9.2.4, "Initial Startup Flow-Induced Vibration Testing of Reactor Internals," is being revised and new proposed license conditions are being added to Part 10 to address the ESBWR steam dryer flow-induced vibration assessment program.
- A new FSAR Section 8.2.1.2.2, "Monitoring of Transformers for Open Circuit," is being added to address the development and implementation of operating, maintenance and testing procedures, and the conduct of training associated with the transformer voltage monitoring system.
- Editorial changes are being made to the Technical Specifications Bases (Part 4).

This information will be incorporated into the next submission of the North Anna Unit 3 COLA, as described in the enclosure.

D089
NRO

Please contact Regina Borsh at (804) 273-2247 (regina.borsh@dom.com) if you have questions.

Very truly yours,

Mark D Mitchell

Mark D. Mitchell

Vice President – Generation Construction

Enclosure: COLA Markups for Implementation of DCD Revision 10

Commitments made by this letter:

This information will be incorporated into the next submission of the North Anna Unit 3 COLA, as described in the enclosure.

COMMONWEALTH OF VIRGINIA

COUNTY OF HENRICO

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Mark D. Mitchell, who is Vice President – Generation Construction of Virginia Electric and Power Company (Dominion Virginia Power). He has affirmed before me that he is duly authorized to execute and file the foregoing document on behalf of the Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 10TH day of FEBRUARY, 2014

My registration number is 253183 and my

Commission expires: SEPTEMBER 30, 2016

Kathy W Prokopis
Notary Public

KATHY W. PROKOPIS NOTARY PUBLIC MY COMMISSION NUMBER 253183 COMMONWEALTH OF VIRGINIA

cc: U. S. Nuclear Regulatory Commission, Region II
C. P. Patel, NRC
T. S. Dozier, NRC
G. J. Kolcum, NRC

ENCLOSURE

COLA Markups for the Implementation

of DCD Revision 10

1.1.1.5 Proprietary and Security-Related Sensitive Unclassified Non-Safeguards Information (SUNSI)

Proprietary information and SUNSI[1] is withheld from public disclosure and therefore not included in the public version of the FSAR. SUNSI included in the non-public version of the FSAR is appropriately indicated.

1.1.1.6 Acronyms

In addition to the summary list of acronyms in the FSAR frontmatter, acronyms are defined at their first occurrence in FSAR text.

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 Section 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. 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

NAPS SUP 1.10-1

Table 1.10-201 Summary of FSAR Sections Where DCD COL Items Are Addressed

Item No.	Subject/Description of Item	FSAR Section
8.2.4-5-A	Protective Relaying	~~8.2.1.2.2~~ 8.2.1.2.3
8.2.4-6-A	Switchyard DC Power	8.2.1.2.1
8.2.4-7-A	Switchyard AC Power	8.2.1.2.1
8.2.4-8-A	Switchyard Transformer Protection	8.2.1.2.1
8.2.4-9-A	Stability and Reliability of the Offsite Transmission Power Systems	8.2.2.1
8.2.4-10-A	Interface Requirements	8.2.1.1
8.3.4-1-A	Safety Related Battery Float and Equalizing Voltage values	8.3.2.1.1
8.3.4-2-A	Identification and Monitoring of Underground or Inaccessible Power and Control Cables to the PSWS and DG Fuel Oil Transfer System Equipment That Have Accident Mitigating Functions.	8.3.3.2
8A.2.3-1-A	Cathodic Protection System	8A.2.1
9.1-4-A	Fuel Handling Operations	9.1.1.7, 9.1.4.13, 9.1.4.18, and 9.1.4.19
9.1-5-A	Handling of Heavy Loads	9.1.5.6, 9.1.5.8, and 9.1.5.9
9.2.1-1-A	Material Selection	9.2.1.2
9.2.5-1-A	Post Seven Day Makeup to UHS	9.2.5
9.3.2-1-A	Post-Accident Sampling Program	9.3.2.2
9.3.9-1-A	Implementation of Hydrogen Water Chemistry	9.3.9
9.3.9-2-A	Hydrogen and Oxygen Storage and Supply	9.3.9.2
9.3.10-1-A	Oxygen Storage Facility	9.3.10.2
9.3.11-1-A	Determine Need for Zinc Injection System	9.3.11 and 9.3.11.1
9.3.11-2-A	Provide System Description for Zinc Injection System	9.3.11.2 and 9.3.11.4
9.5.1-1-A	Secondary Firewater Storage Source	9.5.1.4
9.5.1-2-A	Secondary Firewater Capacity	9.5.1.4

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.

CWR COL 3.9.9-1-A

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

- ~~NEDE-33259P, "ESBWR 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"~~

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

1. ~~A valid prototype for the Unit 3 reactor internals does not exist. Under this scenario, Unit 3 reactor internals is classified as a prototype per Regulatory Guide 1.20.~~

2. ~~A valid prototype for Unit 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 Unit 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, Unit 3 reactor internals will be classified as non-prototype category I. If any changes to classification for Unit 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.~~

~~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 Unit 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.~~

~~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.~~

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

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

   b. A valid prototype for Unit 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 Unit 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, Unit 3 reactor internals will be classified as non-prototype Category I. If a change to the classification for Unit 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.

2. 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. Section 10.2 of NEDE-33313P provides four elements of a steam dryer Comprehensive Vibration Assessment Program that must be addressed. The following describes the approach for the steam dryer Comprehensive Vibration Assessment Program elements, consistent with RG 1.20 and Section 10.2 of NEDE-33313P:

   a. 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.

   b. The detailed design of the steam dryer will follow the methodology described in DCD Appendix 3L and the incorporated engineering reports. As described in NEDE-33313P, Section 10.2(b), an example of a steam dryer predictive 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. The final detailed design of the ESBWR steam dryer has not yet been completed. Therefore, the example of an as-designed steam dryer that has been subjected to the

predictive 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.

c. The startup program and associated steam dryer 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).

d. Periodic steam dryer inspection during refueling outages is as described in NEDE-33313P, Section 10.2(d), and associated license conditions.

3. Summary of Reactor Internals Vibration Assessment Program

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 Unit 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 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.

Transmission Lines	Rated Current at 100°F
500 kV	3954A
230 kV	2190A

Bus	Rated Current at 100°F	Short Circuit Current (Bus Bracing Limit)
500 kV	3891A	50 kA
230 kV	2750A	40 kA

8.2.1.2.2 Monitoring of Transformers for Open Circuit

Add the following to the end of the section.

NAPS SUP 8.2-1

Plant operating procedures associated with the monitoring system, including off-normal operating procedures, will be developed in accordance with Section 13.5.2.1 at least six months prior to fuel load.

Maintenance and testing procedures associated with the monitoring system, including calibration and setpoint determination procedures, will be developed in accordance with Section 13.5.2.2.6.1 prior to fuel load.

Control Room operator and maintenance technician training associated with the operation and maintenance of the monitoring system will be developed in accordance with Section 13.2.1 for reactor operators and Section 13.2.2 13.2.2 for non-licensed plant staff. Training will be completed prior to fuel load.

NAPS COL 8.2.4-5-A

8.2.1.2.3 Protective Relaying

The 500 kV transmission lines are protected with redundant high-speed relay schemes with re-closing and communication equipment to minimize line outages. The 500 kV switchyard buses have redundant bus differential protection using separate and independent current and control circuits. Generating unit tie-lines and auxiliary transformer underground cable circuits are protected with redundant high-speed relay schemes. [Transformers 1, 2, 3, 5, and 6 are protected with sudden pressure relays and differential relays.] SOF-138

Dominion is responsible for engineering, constructing, operating, and maintaining its electric transmission system, and for interfacing with PJM, the Regional Transmission Organization (RTO). Dominion's responsibility includes designing, maintaining, and operating all switchyard protective

Replace the second sentence of the last paragraph with the following.

NAPS SUP 14.2-4 However, due to insufficient heat loads during the preoperational test phase, the heat exchanger and the AHS performance verification is deferred until the startup phase.

14.2.8.2.18 Plant Service Water System Performance Test

Purpose

Replace the first paragraph with the following.

NAPS SUP 14.2-5 The objective of this test is to verify performance of the PSWS including the AHS along with the RCCWS, and the TCCWS under expected reactor power operation load conditions.

Description

Replace the second sentence with the following.

NAPS SUP 14.2-5 Pertinent parameters shall be monitored in order to provide a verification of proper system flow balancing and heat exchanger and AHS performance under near design or special conditions, as appropriate.

14.2.9 Site-Specific Preoperational and Startup Tests

Replace the second and third paragraphs with the following.

NAPS COL 14.2-5-A This section describes the site specific pre-operational and initial startup tests not addressed in DCD Section 14.2.8.

NAPS COL 14.2-6-A Specific testing to be performed and the applicable acceptance criteria for each preoperational and startup test are documented in test procedures to be made available to the NRC approximately 60 days prior to their intended use for preoperational tests, and not less than 60 days prior to scheduled fuel load for initial startup tests, or as otherwise specified in license conditions. Site-specific preoperational tests are in accordance with the system specifications and associated equipment specifications for equipment in those systems provided by the licensee that are not part of the standard plant described in DCD Section 14.2.8. The tests demonstrate that the installed equipment and systems perform within the limits of these specifications.

BASES

BACKGROUND (continued)

The predicted core reactivity, as represented by k-effective (k_{eff}), is calculated by a 3D core simulator code as a function of cycle exposure. This calculation is performed for projected operating states and conditions throughout the cycle. The monitored k_{eff} is calculated by the core 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.

Reactivity Anomalies ~~satisfies~~ 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

BASES

LCO (continued)

times and therefore an inoperable accumulator does not immediately require declaring a control rod inoperable.

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.

STD COL 16.0-1-A ~~3.1.3-1~~

A.1, A.2, and A.3~~, and A.4~~

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

BASES

ACTIONS (continued)

bypassed in the Rod Control and Information System (RC&IS) to allow continued operation. 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.

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."

STD COL 16.0-1-A ~~3.1.3-1~~

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.

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

BASES

ACTIONS (continued)

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.

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.3~~ 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.

STD COL 16.0-1-A ~~3.1.3-1~~

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 control rods are fully inserted within 3 hours and disarmed (however, they do not need to be isolated from scram). Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed to prevent inadvertent withdrawal during subsequent operations. The control rods can be disarmed by bypassing the rod in the RC&IS or disconnecting its power supply. Required Action C.1 is modified by a Note that allows control rods to be bypassed in the RC&IS if required to allow insertion of the inoperable control rods and continued operation. SR 3.3.2.1.9 provides

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").

Control Rod Scram Accumulators ~~satisfies~~ satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

REFERENCES None.

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) 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).

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

ECCS actuation supports OPERABILITY of the ECCS Instrumentation, "LCO 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation" and therefore is required to be OPERABLE. This Specification addresses OPERABILITY of the ECCS actuation circuitry from the outputs of the DTM functions through the VLU functions, the timers, and the 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.

Although there are four divisions of ECCS actuation for each function, only three ECCS 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 single-failure criterion is met with three OPERABLE ECCS actuation divisions, and because each ECCS actuation division is associated with and receives power from only one of the four electrical divisions.

1. Automatic Depressurization System (ADS)

The ADS actuation divisions receive input from the Reactor Vessel Level - Low, Level 1.0 signal sustained for 10 seconds, or from the Drywell Pressure - High signal sustained for 60 minutes. ADS actuation is required to be OPERABLE in MODES 1, 2, 3, and 4, consistent with the requirements of LCO 3.5.1, "Automatic Depressurization System (ADS) - Operating." ADS actuation is required to be OPERABLE in MODE 5, and in MODE 6 prior to removal of the reactor pressure vessel head, consistent with the requirements of LCO 3.5.3, "Gravity-Driven Cooling System (GDCS) – Shutdown." Three actuation divisions are required to be OPERABLE to ensure that no single actuation failure can preclude the actuation function.

2. Gravity-Driven Cooling System (GDCS) Injection Lines

The GDCS injection line actuation divisions receive input from the Reactor Vessel Level - Low, Level 1.0 signal sustained for 10 seconds, or from the Drywell Pressure - High signal sustained for 60 minutes. GDCS injection line actuation is required to be OPERABLE in MODES 1, 2, 3, and 4, consistent with the requirements of LCO 3.5.2, "Gravity-Driven Cooling System (GDCS) - Operating." GDCS

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

injection line actuation is required to be OPERABLE in MODES 5 and 6, except with the buffer pool gate removed and water level ≥ 7.01 meters (23.0 feet) over the top of the reactor pressure vessel flange, consistent with the requirements of LCO 3.5.3, "Gravity-Driven Cooling System (GDCS) - Shutdown." Three actuation divisions are required to be OPERABLE to ensure that no single actuation failure can preclude the actuation function.

3. GDCS Equalizing Lines

The GDCS equalizing line actuation divisions receive input from the following instrumentation: Reactor Vessel Level - Low, Level 1.0 signal sustained for 10 seconds and Reactor Vessel Level - Low, Level 0.5. GDCS equalizing line actuation is required to be OPERABLE in MODES 1, 2, 3, and 4, consistent with the requirements of LCO 3.5.2, "Gravity-Driven Cooling System (GDCS) - Operating." GDCS equalizing line actuation is required to be OPERABLE in MODES 5 and 6, except with the buffer pool gate removed and water level ≥ 7.01 meters (23.0 feet) over the top of the reactor pressure vessel flange, consistent with the requirements of LCO 3.5.3, "Gravity-Driven Cooling System (GDCS) - Shutdown." Three actuation divisions are required to be OPERABLE to ensure that no single actuation failure can preclude that actuation function.

4. Standby Liquid Control (SLC)

The SLC actuation divisions receive inputs from the Reactor Vessel Level - Low, Level 1.0 signal sustained for 10 seconds. SLC actuation is required to be OPERABLE in MODES 1, 2, 3, and 4 consistent with the requirements of LCO 3.1.7, "Standby Liquid Control (SLC) System." Three actuation divisions are required to be OPERABLE to ensure that no single actuation failure can preclude that actuation function.

ACTIONS

A Note has been provided to modify the ACTIONS related to ECCS divisions of actuation logic. 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

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

NAPS COL 16.0-1-A
3.3.7.2-1

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)."

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 ~~APPLICABLE~~ APPLICABILITY

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 ~~APPLICABLE~~ APPLICABILITY (continued)

CRHAVS actuation supports OPERABILITY of the CRHAVS Instrumentation, LCO 3.3.7.1, "Control Room Habitability Area (CRHA) Heating, Ventilation, and Air Conditioning (HVAC) Subsystem (CRHAVS) Instrumentation," and therefore is required to be OPERABLE. This Specification addresses OPERABILITY of the CRHAVS actuation circuitry from the outputs of the DTM functions through the LDs, which covers the VLU functions and the LDs associated with the CRHA isolation dampers, CRHAVS Emergency Filtration Unit (EFU) fans and isolation dampers, and Nonsafety-Related Distributed Control and Information System (N-DCIS) electrical load breakers, and other nonsafety-related electrical loads in the CRHA.

Although there are four divisions of CRHAVS actuation, only three CRHAVS actuation divisions 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 single-failure criterion is met with three OPERABLE CRHAVS actuation divisions, and because each CRHAVS actuation division is associated with and receives power from only one of the four electrical divisions.

In MODES 1, 2, 3, and 4 the CRHAVS must be OPERABLE 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, maintaining the CRHAVS OPERABLE is not required in MODE 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).

ACTIONS

A.1

Condition A exists when one required CRHAVS actuation division is inoperable. In this Condition, CRHAVS actuation still maintains actuation trip capability, but cannot accommodate a single failure. The 12-hour Completion Time is acceptable based on engineering judgment considering the diversity of sensors available to provide trip signals, the redundancy of the CRHAVS actuation design, and the low

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).

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).

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).

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

A.1

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.

B.1 and B.2

If the Required Action and Completion Time of Condition A are not met, the plant 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 36 hours. The Completion Time is reasonable, based on

BASES

BACKGROUND (continued)

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 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 ~~satisfies~~ satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The SCRRI and SRI functions are required to be OPERABLE to limit decrease in MCPR within acceptable limits during events that cause rapid increase in core reactivity, such that the Fuel Cladding Integrity Safety Limit (FCISL) is not exceeded.

STD COL 16.0-1-A 3.7.6-1

BASES

ACTIONS (continued)

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.

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 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.

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 ~~satisfies~~ 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.

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.

When the reactor mode switch is in the Shutdown position, a control rod block (LCO 3.3.2.1, "Control Rod Block Instrumentation") ensures control rod withdrawal cannot occur simultaneously with in-vessel fuel movement. In

BASES

APPLICABLE SAFETY ANALYSES (continued)

control rod pair withdrawn, the core will remain subcritical, thereby preventing any prompt critical excursion.

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.

Control rod withdrawal must be immediately suspended, and action must be immediately initiated to fully insert all insertable control rods in core cells containing one or more

(2) At least once every 10 years, each explosively actuated valve shall be disassembled for internal examination of the valve and actuator to verify the operational readiness of the valve assembly and the integrity of individual components and to remove any foreign material, fluid, or corrosion. The examination schedule shall provide for each valve design used for explosively actuated valves at the facility to be included among the explosively actuated valves to be disassembled and examined every 2 years. Corrective action shall be taken to resolve any deficiencies identified during the examination with post-maintenance testing conducted that satisfies the PST requirements.

(3) For explosively actuated valves selected for test sampling every 2 years in accordance with the ASME OM Code, the operational readiness of the actuation logic and associated electrical circuits shall be verified for each sampled explosively actuated valve following removal of its charge. This must include confirmation that sufficient electrical parameters (voltage, current, resistance) are available for each valve actuation circuit. Corrective action shall be taken to resolve any deficiencies identified in the actuation logic or associated electrical circuits.

(4) For explosively actuated valves selected for test sampling every 2 years in accordance with the ASME OM Code, the sampling must select at least one explosively actuated valve from each redundant safety train. Each sampled pyrotechnic charge shall be tested in the valve or a qualified test fixture to confirm the capability of the charge to provide the necessary motive force to operate the valve to perform its intended function without damage to the valve body or connected piping. Corrective action shall be taken to resolve any deficiencies identified in the capability of a pyrotechnic charge in accordance with the PST requirements.

This license condition supplements the current requirements in the ASME OM Code for explosively actuated valves, and sets forth requirements for preservice testing and operational surveillance, as well as any necessary condition. The license condition will expire either when (1) the license condition is incorporated into the Unit 3 Inservice Testing (IST) program; or (2) the updated ASME OM Code requirements for squib valves in new reactors (i.e., plants receiving a construction permit, or a combined license for construction and operation, after January 1, 2000), as accepted by the NRC in 10 CFR 50.55a, are incorporated into the Unit 3 IST program. For the purpose of satisfying the license condition, the licensee retains the option of including in its IST program either the requirements stated in this condition, or including updated ASME OM Code requirements.

3.10 Steam Dryer License Conditions

The licensee shall implement the following license conditions using supporting information in GE Hitachi Nuclear Energy Reports NEDE-33312P, "ESBWR Steam Dryer Acoustic Load Definition,"

Revision 5, December 2013, and NEDE-33313P, "ESBWR Steam Dryer Structural Evaluation," Revision 5, December 2013.

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

1.b. Power Ascension Test (PAT) procedures for the steam dryer testing shall be provided to NRC inspectors no later than 10 days before initial fuel load. The PAT procedures shall 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 power ascension shall be at no more than 75 percent of full power. At this hold point, the licensee shall complete the actions specified in item 2 of the model license condition specified in paragraph (c) of Section 10.2, "Comprehensive Vibration Program Elements for a COL Applicant," in NEDE-33313-P, Revision 5.

3. Continue power ascension: The licensee shall complete the actions specified in item 3 of the model license condition specified in paragraph (c) of Section 10.2 in NEDE-33313P, Revision 5.

4. Power ascension monitoring: The licensee shall complete the actions specified in item 4 of the model license condition specified in paragraph (c) of Section 10.2 in NEDE-33313P, Revision 5.

5. Flow-induced resonances: The licensee shall complete the actions specified in item 5 of the model license condition specified in paragraph (c) of Section 10.2 in NEDE-33313P, Revision 5.

6. Limit curve modifications: The licensee shall complete the actions specified in item 6 of the model license condition specified in paragraph (c) of Section 10.2 in NEDE-33313P, Revision 5.

7. At the initial hold point and the hold points at approximately 85 and 95 percent power, power ascension shall not proceed for at least 72 hours after making the steam dryer data analysis and results available to the NRC by facsimile or electronic transmission to the NRC project manager.

8. During the Power Maneuvering in the Feedwater Temperature Operating Domain testing, pressures, strains, and accelerations shall 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 shall be included in the stress analysis report described in license condition 3.10.9 (below).

9. Full power achievement: The licensee shall complete the actions specified in item 9 of the model license condition specified in paragraph (c) of Section 10.2 in NEDE-33313P, Revision 5.

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 shall 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 shall 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 shall be provided to the NRC within 180 days following startup from the second refueling outage.