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July 25, 2005

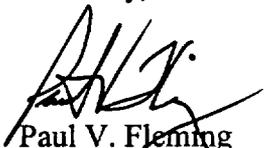
U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Technical Specification Bases Changes
Cooper Nuclear Station, Docket No. 50-298, DPR-46

The purpose of this letter is to provide changes to the Cooper Nuclear Station (CNS) Technical Specification Bases implemented without prior Nuclear Regulatory Commission approval. In accordance with the requirements of CNS Technical Specification 5.5.10.d, these changes are provided on a frequency consistent with 10 CFR 50.71(e). The enclosed Bases changes are for the time period from August 16, 2003, through May 31, 2005. Also enclosed are filing instructions and an updated List of Effective Pages for the CNS Technical Specification Bases.

If you have any questions regarding this submittal, please contact me at (402) 825-2774.

Sincerely,



Paul V. Fleming
Licensing Manager

/lb

Enclosure

cc: Regional Administrator, w/enclosure
USNRC - Region IV

Senior Project Manager, w/enclosure
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector, w/enclosure
USNRC

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FILING INSTRUCTIONS

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B 3.8-30	0	B 3.8-74	0
B 3.8-31	0	B 3.8-75	0
B 3.8-32	0		
B 3.8-33	10/21/04	B 3.9-1	12/18/03
B 3.8-34	10/21/04	B 3.9-2	0
B 3.8-35	10/21/04	B 3.9-3	0
B 3.8-36	10/21/04	B 3.9-4	12/18/03
B 3.8-37	1	B 3.9-5	12/18/03
B 3.8-38	0	B 3.9-6	6/28/01
B 3.8-39	0	B 3.9-7	6/28/01
B 3.8-40	1	B 3.9-8	12/18/03
B 3.8-41	0	B 3.9-9	12/18/03
B 3.8-42	0	B 3.9-10	0
B 3.8-43	0	B 3.9-11	12/18/03
B 3.8-44	0	B 3.9-12	12/18/03
B 3.8-45	0	B 3.9-13	0
B 3.8-46	0	B 3.9-14	0
B 3.8-47	0	B 3.9-15	12/18/03
B 3.8-48	0	B 3.9-16	12/18/03
B 3.8-49	0	B 3.9-17	0
B 3.8-50	07/07/03	B 3.9-18	12/18/03
B 3.8-51	0	B 3.9-19	11/04/01
B 3.8-52	0	B 3.9-20	0
B 3.8-53	0	B 3.9-21	0
B 3.8-54	0	B 3.9-22	0
B 3.8-55	0	B 3.9-23	0
B 3.8-56	0	B 3.9-24	0
B 3.8-57	0	B 3.9-25	0
B 3.8-58	0	B 3.9-26	0
B 3.8-59	0	B 3.9-27	0
B 3.8-60	0	B 3.9-28	0
B 3.8-61	0	B 3.9-29	0
B 3.8-62	0	B 3.9-30	0
B 3.8-63	0		
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B 3.10-9	0		
B 3.10-10	0		
B 3.10-11	0		
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B 3.10-13	0		
B 3.10-14	0		
B 3.10-15	0		
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B 3.10-34	0		
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B 3.10-36	0		
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B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

USAR, Appendix F (Ref. 1) establishes, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and abnormal operational transients.

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2 for General Electric Company (GE) fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during abnormal operational transients, at least 99.9% of the fuel rods in the core do not experience transition boiling.

(continued)

BASES

**SAFETY LIMIT
VIOLATIONS**

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 3). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. USAR, Appendix F, Section F-2.2.1.
 2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (Revision specified in the COLR).
 3. 10 CFR 100.
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B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to the USAR, Appendix F (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

During normal operation and abnormal operational transients, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

(continued)

BASES (continued)

**SAFETY LIMIT
VIOLATIONS**

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. USAR, Appendix F, Section F-2.6.1.
 2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
 3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
 4. 10 CFR 100.
 5. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, Addenda winter of 1966.
 6. ASME, USAS, Nuclear Power Piping Code, Section B31.1, 1967 Edition.
 7. ASME, Boiler and Pressure Vessel Code, Section III, 1983 Edition.
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BASES

- REFERENCES
1. USAR, Appendix F, Section F-2.5.
 2. USAR, Section XIV-6.0.
 3. NEDE-24011-P-A-US, "General Electric Standard Application for Reactor Fuel," Supplement for United States (Revision specified in the COLR).
 4. USAR, Section XIV-5.3.3.
 5. USAR, Section XIV-5.3.4.
 6. 10 CFR 50.36(c)(2)(ii).
 7. Deleted
 8. USAR, Section III-6.
 9. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1 (Revision specified in the COLR).
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

BASES

BACKGROUND

In accordance with USAR, Appendix F (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under the most reactive conditions and acceptable fuel design limits are not exceeded during normal operation and abnormal operational transients. Therefore, Reactivity Anomalies are used as a measure of the predicted versus measured core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that the Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity anomaly could be the result of unanticipated changes in fuel reactivity or control rod worth or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM requirements (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in assuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers, producing zero net reactivity.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel loaded in the previous cycles provide excess positive reactivity beyond that required to sustain steady state operation at the beginning of cycle (BOC). When the reactor is critical at RTP and operating moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, and whatever neutron poisons (mainly xenon and samarium) are present in the fuel. The predicted core reactivity, as represented by control rod density, 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 core reactivity is determined from control rod densities for actual plant conditions and is then compared to the predicted value for the cycle exposure.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.2.1 (continued)

For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (no control rod movement or core flow changes) at $\geq 75\%$ RTP have been obtained. The 1000 MWD/T Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1. The tests performed at this Frequency also use base data obtained during the first test of the specific fuel cycle.

REFERENCES

1. USAR, Appendix F, Section F-2.5.
 2. USAR, Section XIV.
 3. 10 CFR 50.36(c)(2)(ii).
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Control Rod OPERABILITY

BASES

BACKGROUND Control rods are components of the Control Rod Drive (CRD) System, which is the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes to ensure under conditions of normal operation, including abnormal operational transients, that specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the criteria specified in Reference 1.

The CRD System consists of 137 locking piston control rod drive mechanisms (CRDMs) and a hydraulic control unit for each drive mechanism. The locking piston type CRDM is a double acting hydraulic piston, which uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion.

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses of References 2, 3, and 4.

APPLICABLE SAFETY ANALYSES The analytical methods and assumptions used in the evaluations involving control rods are presented in References 2, 3, and 4. The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical and for limiting the potential effects of reactivity insertion events caused by malfunctions in the CRD System.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.3.5

Coupling verification is performed to ensure the control rod is connected to the CRDM and will perform its intended function when necessary. The Surveillance requires verifying a control rod does not go to the withdrawn overtravel position. The overtravel position feature provides a positive check on the coupling integrity since only an uncoupled CRD can reach the overtravel position. The verification is required to be performed any time a control rod is withdrawn to the "full out" position (notch position 48) or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This includes control rods inserted one notch and then returned to the "full out" position during the performance of SR 3.1.3.2. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved and operating experience related to uncoupling events.

REFERENCES

1. USAR, Appendix F, Section F-2.5.
2. USAR, Section XIV-5.3.1
3. USAR, Section XIV-6.2.
4. USAR, Appendix G.
5. 10 CFR 50.36(c)(2)(ii).
6. NEDO-21231, "Banked Position Withdrawal Sequence," Section 7.2, January 1977.

BASES

- REFERENCES
1. USAR, Appendix F, Section F-2.2.1.
 2. USAR, Section III-5.
 3. USAR, Section VII-2.0.
 4. USAR, Section XIV.
 5. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, Revision specified in the COLR.
 6. 10 CFR 50.36(c)(2)(ii).
 7. Letter from R.F. Janecek (BWROG) to R.W. Starostecki (NRC), "BWR Owners Group Revised Reactivity Control System Technical Specifications," BWROG-8754, September 17, 1987.
 8. Technical Requirements Manual.
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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Control rod patterns analyzed in Reference 1 follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to 10% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. Generic analysis of the BPWS (Ref. 1) has demonstrated that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS mode of operation. The generic BPWS analysis (Ref. 8) also evaluates the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods.

When performing a shutdown of the plant, an optional BPWS control rod sequence (Ref. 10) may be used provided that all withdrawn control rods have been confirmed to be coupled. The rods may be inserted without the need to stop at intermediate positions since the possibility of a CRDA is eliminated by the confirmation that withdrawn control rods are coupled. When using the Reference 10 control rod sequence for shutdown, the rod worth minimizer may be reprogrammed to enforce the requirements of the improved BPWS control rod insertion, or may be bypassed and the improved BPWS shutdown sequence implemented under LCO 3.3.2.1, Condition D controls.

In order to use the Reference 10 BPWS shutdown process, an extra check is required in order to consider a control rod to be "confirmed" to be coupled. This extra check ensures that no Single Operator Error can result in an incorrect coupling check. For purposes of this shutdown process, the method for confirming that control rods are coupled varies depending on the position of the control rod in the core. Details on this coupling confirmation requirement are provided in Reference 10. If the requirements for use of the BPWS control rod insertion process contained in Reference 10 are followed, the plant is considered to be in compliance with BPWS requirements, as required by LCO 3.1.6.

Rod pattern control satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 9).

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.

BASES (continued)

APPLICABILITY In MODES 1 and 2, when THERMAL POWER is $\leq 10\%$ RTP, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is $> 10\%$ RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 2). In MODES 3, 4, and 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.

ACTIONS

A.1 and A.2

With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence, actions may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, or a power reduction to $\leq 10\%$ RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to eight, to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence.

Required Action A.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator (Reactor Operator or Senior Reactor Operator) or by a qualified member of the technical staff. This ensures that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not considered inoperable except as required by Required Action A.2. The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod

BASES

ACTIONS

B.1 and B.2 (continued)

worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator (Reactor Operator or Senior Reactor Operator) or by a qualified member of the technical staff.

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE
REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at $\leq 10\%$ RTP.

REFERENCES

1. NEDE-24011-P-A-US, "General Electric Standard Application for Reactor Fuel, Supplement for United States," Section 2.2.3.1 (Revision specified in the COLR).
2. "Modifications to the Requirements for Control Rod Drop Accident Mitigating System," BWR Owners Group, July 1986.
3. NUREG-0979, Section 4.2.1.3.2, April 1983.
4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.
5. 10 CFR 100.

BASES

REFERENCES

(continued)

6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.
 7. ASME, Boiler and Pressure Vessel Code.
 8. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.
 9. 10 CFR 50.36(c)(2)(ii).
 10. NEDO 33091, Revision 2, "Improved BPWS Control Rod Insertion Process," April 2003.
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BASES

APPLICABLE SAFETY ANALYSES
LCO, and
APPLICABILITY

2. Rod Worth Minimizer (continued)

analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in References 5 and 6. The BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the BPWS are specified in LCO 3.1.6, "Rod Pattern Control."

When performing a shutdown of the plant, an optional BPWS control rod sequence (Ref. 7) may be used if the coupling of each withdrawn control rod has been confirmed. The rods may be inserted without the need to stop at intermediate positions. When using the Reference 7 control rod insertion sequence for shutdown, the rod worth minimizer may be reprogrammed to enforce the requirements of the improved BPWS control rod insertion, or may be bypassed and the improved BPWS shutdown sequence implemented under the controls in Condition D.

The RWM Function satisfies Criterion 3 of Reference 4.

Since the RWM is a system designed to act as a backup to operator control of the rod sequences, only one channel of the RWM is available and required to be OPERABLE (Ref. 7). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the BPWS. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed.

Compliance with the BPWS, and therefore OPERABILITY of the RWM, is required in MODES 1 and 2 when THERMAL POWER is $\leq 10\%$ RTP. When THERMAL POWER is $> 10\%$ RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 5). In MODES 3 and 4, all control rods are required to be inserted into the core; therefore, a CRDA cannot occur. In MODE 5, since only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will be subcritical.

BASES

APPLICABLE
SAFETY ANALYSIS
LCO, AND
APPLICABILITY

3. Reactor Mode Switch-Shutdown Position

During MODES 3 and 4, and during MODE 5 when the reactor mode switch is required to be in the shutdown position, the core is assumed to be subcritical; therefore, no positive reactivity insertion events are analyzed. The Reactor Mode Switch — Shutdown Position control rod withdrawal block ensures that the reactor remains subcritical by blocking control rod withdrawal, thereby preserving the assumptions of the safety analysis.

The Reactor Mode Switch — Shutdown Position Function satisfies Criterion 3 of Reference 4. Two channels are required to be OPERABLE to ensure that no single channel failure will preclude a rod block when required. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on reactor mode switch position.

During shutdown conditions (MODE 3, 4, or 5), no positive reactivity insertion events are analyzed because assumptions are that control rod withdrawal blocks are provided to prevent criticality. Therefore, when the reactor mode switch is in the shutdown position, the control rod withdrawal block is required to be OPERABLE. During MODE 5 with the reactor mode switch in the refueling position, the refuel position one-rod-out interlock (LCO 3.9.2, "Refuel Position One-Rod-Out Interlock") provides the required control rod withdrawal blocks.

ACTIONS

A.1

With one RBM channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod block function; however, overall reliability is reduced because a single failure in the remaining OPERABLE channel can result in no control rod block capability for the RBM. For this reason, Required Action A.1 requires restoration of the inoperable channel to OPERABLE status. The Completion Time of 24 hours is based on the low probability of an event occurring coincident with a failure in the remaining OPERABLE channel.

B.1

If Required Action A.1 is not met and the associated Completion Time has expired, the inoperable channel must be placed in trip within 1 hour. If both RBM channels are inoperable, the RBM is not capable of performing its intended function; thus, one channel must also be placed

BASES

ACTIONS

B.1 (continued)

in trip. This initiates a control rod withdrawal block, thereby ensuring that the RBM function is met.

The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping of inoperable channels.

C.1, C.2.1.1, C.2.1.2, and C.2.2

With the RWM inoperable during a reactor startup, the operator is still capable of enforcing the prescribed control rod sequence. However, the overall reliability is reduced because a single operator error can result in violating the control rod sequence. Therefore, control rod movement must be immediately suspended except by scram. Alternatively, startup may continue if at least 12 control rods have already been withdrawn, or a reactor startup with an inoperable RWM during withdrawal of one or more of the first 12 rods was not performed in the last (current) calendar year. These requirements minimize the number of reactor startups initiated with the RWM inoperable. Required Actions C.2.1.1 and C.2.1.2 require verification of these conditions by review of plant logs and control room indications. Once Required Action C.2.1.1 or C.2.1.2 is satisfactorily completed, control rod withdrawal may proceed in accordance with the restrictions imposed by Required Action C.2.2. Required Action C.2.2 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff.

The RWM may be bypassed under these conditions to allow continued operations. In addition, Required Actions of LCO 3.1.3 and LCO 3.1.6 may require bypassing the RWM, during which time the RWM must be considered inoperable with Condition C entered and its Required Actions taken.

BASES

ACTIONS
(continued)

D.1

With the RWM inoperable during a reactor shutdown, the operator is still capable of enforcing the prescribed control rod sequence. Required Action D.1 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff. The RWM may be bypassed under these conditions to allow the reactor shutdown to continue.

E.1 and E.2

With one Reactor Mode Switch — Shutdown Position control rod withdrawal block channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod withdrawal block function. However, since the Required Actions are consistent with the normal action of an OPERABLE Reactor Mode Switch — Shutdown Position Function (i.e., maintaining all control rods inserted), there is no distinction between having one or two channels inoperable.

In both cases (one or both channels inoperable), suspending all control rod withdrawal and initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies will ensure that the core is subcritical with adequate SDM ensured by LCO 3.1.1. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are therefore not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

SURVEILLANCE
REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are modified by a second Note to indicate that when an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and

BASES

SURVEILLANCE REQUIREMENTS (continued)

Required Actions taken. This Note is based on the reliability analysis (Ref. 9) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the channel will perform the intended function. It includes the Reactor Manual Control System input. It also includes the local alarm lights representing upscale and downscale trips, but no rod block will be produced at this time. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of 92 days is based on reliability analyses (Ref. 10).

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the system will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The CHANNEL FUNCTIONAL TEST for the RWM includes performing the RWM computer on line diagnostic test satisfactorily, attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs. For SR 3.3.2.1.2, the CHANNEL FUNCTIONAL TEST also includes attempting to select a control rod not in compliance with the prescribed sequence and verifying a selection error occurs. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn in MODE 2. As noted, SR 3.3.2.1.3 is not required to be performed until 1 hour after THERMAL POWER is \leq 10% RTP in MODE 1. This allows

BASES

SURVEILLANCE REQUIREMENTS (continued)

entry into MODE 2 for SR 3.3.2.1.2, and entry into MODE 1 when THERMAL POWER is $\leq 10\%$ RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 10).

SR 3.3.2.1.4

The RBM power range setpoints control the enforcement of the appropriate upscale trips over the proper core thermal power range of the Applicability Notes (a), (b), (c), (d), and (e) of ITS Table 3.3.2.1-1. The RBM Upscale Trip Function setpoints are automatically varied as a function of power. Three Allowable Values are specified in the COLR as denoted in Table 3.3.2.1-1, each within a specific power range. The power at which the control rod block Allowable Values automatically change are based on the reference APRM signal's input to each RBM channel. Below the minimum power setpoint of 27.5% RTP or when a peripheral control rod is selected, the RBM is automatically bypassed. These power Allowable Values must be verified periodically by determining that the power level setpoints are less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.8. The 184 day Frequency is based on the actual trip setpoint methodology utilized for these channels.

SR 3.3.2.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

BASES

SURVEILLANCE REQUIREMENTS (continued)

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.8.

The Frequency is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.2.1.6

The RWM is automatically bypassed when power is above a specified value. The power level is determined from feedwater flow and steam flow signals. The setpoint where the automatic bypass feature is unbypassed must be verified periodically to be > 10% RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on the trip setpoint methodology utilized for the low power setpoint channel.

SR 3.3.2.1.7

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch — Shutdown Position Function to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch — Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into MODES 3 and 4 if the 18 month Frequency

BASES

SURVEILLANCE REQUIREMENTS (continued)

is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs.

The 18 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 these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.2.1.8

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

BASES

REFERENCES

1. USAR, Section VII-7.
 2. USAR, Section VII-16.3.3.
 3. NEDC-31892P, "Extended Load Line Limit and ARTS Improvement Program Analyses for Cooper Nuclear Station," Rev. 1, May 1991.
 4. 10 CFR 50.36(c)(2)(ii).
 5. USAR, Section XIV-6.2.
 6. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.
 7. NEDO 33091, Revision 2, "Improved BPWS Control Rod Insertion Process," April 2003.
 8. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 2987.
 9. GENE-770-06-1, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.
 10. NEDC-30851-P-A, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.
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BASES

BACKGROUND

3. 4. High Pressure Coolant Injection System Isolation and
Reactor Core Isolation Cooling System Isolation (continued)

The HPCI Steam Line Space Temperature-High Function receives input from 32 bimetallic temperature switches physically located along and in the vicinity of the HPCI steam line. Additionally, 8 temperature switches located along and in the vicinity of the RHR steam condensing mode steam lines input into this Function. These 40 switches are located in groups of eight (8). The 32 HPCI steam line switches cover four locations; RHR injection valve room, torus area west, SW quadrant, and the HPCI pump room. The 8 RHR steam condensing line switches are located in torus area NW. For each location, four switches input into trip system A, the other four switches input to trip system B. Each set of four switches is arranged in a one-of-two taken twice logic trip system. One trip system isolates the HPCI steam line inboard isolation valves and the other trip system isolates the HPCI steam line outboard valves. For purposes of this specification, each temperature switch is considered a "channel".

The RCIC Steam Line Space Temperature-High Function receives input from 16 bimetallic temperature switches located along and in the vicinity of the RCIC steam line; 8 switches are located in the torus area NE, the remaining 8 are located in the NE quadrant RCIC pump room. For each location, four switches input to trip system A, the other four switches input to trip system B. Each set of four switches is arranged in a one-out-of-two taken twice logic trip system. One trip system isolates the RCIC Steam Line Inboard Isolation Valve, and the other trip system isolates the RCIC Steam Line Outboard Isolation Valve. For purposes of this specification, each temperature switch is considered a "channel".

The HPCI and RCIC Steam Line Flow-High Functions, Steam Supply Pressure-Low Functions, and Steam Line Space Temperature-High Functions isolate the associated steam supply. The Functions associated with HPCI close the HPCI pump suction valve from the suppression pool (if the ECST suction valve is open), close the HPCI turbine exhaust line

BASES

BACKGROUND

3. 4. High Pressure Coolant Injection System Isolation and Reactor Core Isolation Cooling System Isolation (continued)

drain pot drain valves, and cause a HPCI turbine trip which closes the HPCI minimum flow valve. The Functions associated with RCIC cause a RCIC turbine trip which closes the RCIC minimum flow valve.

5. Reactor Water Cleanup System Isolation

The Reactor Vessel Water Level — Low Low (Level 2) Isolation Function receives input from four reactor vessel water level channels. The outputs from the reactor vessel water level channels are connected into two one-out-of-two taken twice trip systems. The RWCU Flow — High Function receives input from two channels, with each channel in one trip system using a one-out-of-one logic, with one channel tripping the inboard RWCU isolation valve and one channel tripping the outboard RWCU isolation valve. The RWCU System Space Temperature-High Function receives input from 48 bimetallic temperature switches. These switches are physically located along and in the vicinity of the RWCU system high temperature piping in groups of eight (8). Thus, there are six (6) locations; RWCU HX room NW (RWCU supply line), RWCU pump rooms (2 locations), RWCU HX room (pump discharge line to Regenerative HX), torus area south, and torus area east. For each location, four switches input into trip system A, the other four switches input into trip system B. Each set of four switches is arranged in a one-out-of-two taken twice logic in series with a normally deenergized trip relay. Actuation of the correct combination of two switches will initiate the corresponding trip system. Trip system A isolates the RWCU supply line inboard isolation valve, and trip system B isolates the RWCU supply line outboard isolation valve. For purposes of this specification, each temperature switch is considered a "channel".

The SLC System Isolation Function receives input from two channels (one channel in each trip system), arranged in a one-out-of-one logic. A channel consists of one of the two control room SLC pump start switches which inputs directly into one of the two RWCU isolation logic trip systems. Placing the SLC Pump A control switch to "Start" will isolate the RWCU inboard isolation valve. Placing the

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Reactor Water Cleanup System Isolation

5.a. RWCU Flow - High

The high flow signal is provided to detect a break in the RWCU System. Should the reactor coolant continue to flow out of the break, offsite dose limits may be exceeded. Therefore, isolation of the RWCU System is initiated when high flow is sensed to prevent exceeding offsite doses. This Function is not assumed in any USAR transient or accident analysis, since bounding analyses are performed for large breaks such as MSLBs.

The high RWCU flow signals are initiated from differential pressure switches that are connected to an annubar on the inlet pump suction line of the RWCU System. Two channels of RWCU Flow — High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The RWCU Flow — High Allowable Value ensures that a break of the RWCU piping is detected.

This Function isolates the Group 3 valves, as listed in Reference 1.

5.b. RWCU System Space Temperature - High

RWCU System Space temperatures are provided to detect a leak from the RWCU System. The isolation occurs even when very small leaks have occurred and is diverse to the high flow instrumentation for the hot portions of the RWCU System. If the small leak continues without isolation, offsite dose limits may be reached. Credit for these instruments is not taken in any transient or accident analysis in the USAR, since bounding analyses are performed for large breaks such as recirculation or MSL breaks.

RWCU System Space temperature signals are initiated from temperature switches (channels) located in the vicinity of high temperature RWCU piping. For each physical location of eight channels, only two channels per trip system are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Since the logic configuration for a trip system is one-out-of-two taken twice, the two required OPERABLE channels per trip

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

5.b RWCU System Space Temperature - High (continued)

system must be in different trip strings, i.e., they must be connected such that isolation occurs when both required channels actuate (one of two parallel logic pairs of switch channels in one trip string must trip in combination with the tripping of one of two additional parallel logic switch channels in the other trip string in order to actuate the trip system.

The RWCU System Space Temperature — High Allowable Values are set low enough to detect a leak.

These Functions isolate the Group 3 valves, as listed in Reference 1.

5.c. Standby Liquid Control (SLC) System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 9). RWCU isolation signals from the SLC system actuation are initiated from the two control room SLC pump start signals.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

Two channels of the SLC System Initiation Function are available and are required to be OPERABLE only in MODES 1 and 2, since these are the only MODES where the reactor can be critical, and these MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

This Function isolates the inboard and outboard RWCU suction valves.

5.d. Reactor Vessel Water Level - Low Low (Level 2)

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 2 supports actions to ensure that the fuel

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

5.d Reactor Vessel Water Level - Low Low (Level 2) (continued)

peak cladding temperature remains below the limits of 10 CFR 50.46. The Reactor Vessel Water Level — Low Low (Level 2) Function associated with RWCU isolation is not directly assumed in the USAR safety analyses because the RWCU System line break is bounded by breaks of larger systems (recirculation and MSL breaks are more limiting).

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

The Reactor Vessel Water Level — Low Low (Level 2) Allowable Value was chosen to be the same as the High Pressure Coolant Injection/Reactor Core Isolation Cooling (HPCI/RCIC) Reactor Vessel Water Level — Low Low (Level 2) Allowable Value (LCO 3.3.5.1 and LCO 3.3.5.2), since this could indicate that the capability to cool the fuel may be threatened.

This Function isolates the Group 3 valves, as listed in Reference 1.

Shutdown Cooling System Isolation

6.a. Reactor Pressure - High

The Reactor Pressure — High Function is provided to isolate the shutdown cooling portion of the Residual Heat Removal (RHR) System. This Function is provided only for equipment protection to prevent an intersystem LOCA scenario, and credit for the interlock is not assumed in the accident or transient analysis in the USAR.

The Reactor Pressure — High signals are initiated from two pressure switches that are connected to different taps on a recirculation pump suction line. Two channels of Reactor Pressure — High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. The Function is only required to be OPERABLE in MODES 1, 2, and 3, since these

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis or appropriate documents. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions when SCIVs and the SGT System are required.

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

1. Reactor Vessel Water Level - Low Low (Level 2)

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite dose release. The Reactor Vessel Water Level — Low Low (Level 2) Function is one of the Functions assumed to be OPERABLE and capable of providing isolation and initiation signals. The isolation and initiation systems on Reactor Vessel Water Level — Low Low (Level 2) support actions to ensure that any offsite releases are within the limits calculated in the safety analysis.

Reactor Vessel Water Level — Low Low (Level 2) signals are initiated from level switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Reactor Vessel Water Level-Low Low (Level 2) (continued)

the actual water level (variable leg) in the vessel. Eight channels of Reactor Vessel Water Level — Low Low (Level 2) Function are available and are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level — Low Low (Level 2) Allowable Value was chosen to be the same as the High Pressure Coolant Injection/ Reactor Core Isolation Cooling (HPCI/RCIC) Reactor Vessel Water Level Low Low (Level 2) Allowable Value (LCO 3.3.5.1 and LCO 3.3.5.2) since this could indicate that the capability to cool the fuel is being threatened).

The Reactor Vessel Water Level — Low Low (Level 2) Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the Reactor Coolant System (RCS); thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, this Function is not required. In addition, the Function is also required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) because the capability of isolating potential sources of leakage must be provided to ensure that offsite dose limits are not exceeded if core damage occurs.

2. Drywell Pressure-High

High drywell pressure can indicate a break in the reactor coolant pressure boundary (RCPB). An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite dose release. The isolation on high drywell pressure supports actions to ensure that any offsite releases are within the limits calculated in the safety analysis. The Drywell Pressure — High Function associated with isolation is not assumed in any USAR accident or transient analyses, but will provide an isolation and initiation signal. It is retained for the overall redundancy and diversity of the secondary containment isolation instrumentation as required by the NRC approved licensing basis.

B 3.3 INSTRUMENTATION

B 3.3.7.1 Control Room Emergency Filter (CREF) System Instrumentation

BASES

BACKGROUND The CREF System is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. The instrumentation and controls for the CREF System automatically isolate the normal ventilation intake and initiate action to pressurize the main control room and filter incoming air to minimize the infiltration of radioactive material into the control room environment.

In the event of a loss of coolant accident (LOCA) signal (Reactor Vessel Water Level — Low Low, Level 2 or Drywell Pressure — High) or Reactor Building Ventilation Exhaust Plenum Radiation — High signal, the normal control room inlet supply damper closes and the CREF System is automatically started in the emergency bypass mode. The air drawn in from the outside passes through a high efficiency filter and a charcoal filter in sufficient volume to maintain the control room slightly pressurized with respect to the adjacent areas.

The CREF System instrumentation has two trip systems. Each trip system includes the sensors, relays, and switches necessary to cause initiation of the CREF System. Each trip system receives input from each of the Functions listed above (each sensor sends a signal to both trip systems). The Reactor Vessel Water Level — Low Low, Level 2, Drywell Pressure — High, and Reactor Building Ventilation Exhaust Plenum Radiation — High are each arranged in a one-out-of-two taken twice logic for each trip system. The channels include electronic and electrical equipment (e.g., switches and trip relays) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a CREF System initiation signal to the initiation logic.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ability of the CREF System to maintain the habitability of the control room is explicitly assumed for certain accidents as discussed in the USAR safety analyses (Refs. 1, 2, and 3). CREF System operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents that assume CREF System operation, does not exceed the limits set by GDC 19 of 10 CFR 50, Appendix A.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Reactor Vessel Water Level — Low Low (Level 2)

Low reactor pressure vessel (RPV) water level indicates that the capability of cooling the fuel may be threatened. A low reactor vessel water level could indicate a LOCA and will automatically initiate the CREF System, since this could be a precursor to a potential radiation release and subsequent radiation exposure to control room personnel.

Reactor Vessel Water Level — Low Low (Level 2) signals are initiated from level switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Eight channels of Reactor Vessel Water Level — Low Low (Level 2) Function are available and are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude CREF System initiation.

The Reactor Vessel Water Level — Low Low (Level 2) Allowable Value was chosen to be the same as the Secondary Containment Isolation Allowable Value (LCO 3.3.6.2) to enable initiation of the CREF System at the earliest indication of a breach in the nuclear system process barrier, yet far enough below normal operational levels to avoid spurious initiation.

The Reactor Vessel Water Level — Low Low (Level 2) Function is required to be OPERABLE in MODES 1, 2, and 3, and during operations with a potential for draining the reactor vessel (OPDRVs) to ensure that the Control Room personnel are protected during a LOCA. In MODES 4 and 5 at times other than OPDRVs, the probability of a vessel draindown event resulting in the release of radioactive material to the environment is minimal. Therefore, this Function is not required in other MODES and specified conditions.

2. Drywell Pressure — High

High drywell pressure can indicate a break in the reactor coolant pressure boundary. A high drywell pressure signal could indicate a LOCA and will automatically initiate the CREF System, since this could be a precursor to a potential radiation release and subsequent radiation exposure to control room personnel.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. Drywell Pressure — High (continued)

Drywell Pressure — High signals are initiated from pressure switches that sense drywell pressure. Eight channels of Drywell Pressure — High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude performance of the initiation function. The Drywell Pressure — High Allowable Value was chosen to be the same as the ECCS Drywell Pressure — High Function Allowable Value (LCO 3.3.5.1).

The Drywell Pressure — High Function is required to be OPERABLE in MODES 1, 2, and 3 to ensure that control room personnel are protected in the event of a LOCA. In MODES 4 and 5, the Drywell Pressure — High Function is not required since there is insufficient energy in the reactor to pressurize the drywell to the Drywell Pressure — High setpoint.

3. Reactor Building Ventilation Exhaust Plenum Radiation — High

High radiation in the refueling floor area could be the result of a fuel handling accident. A refueling floor high radiation signal will automatically initiate the CREF System, since this radiation release could result in radiation exposure to control room personnel.

The Reactor Building Exhaust Plenum Radiation — High signals are initiated from radiation detectors that are located such that they can monitor the radioactivity of gas flowing through the reactor building exhaust plenum. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel in each trip system. Four channels of Reactor Building Ventilation Exhaust Plenum Radiation — High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the CREF System initiation. The Allowable Value was chosen to promptly detect gross failure of the fuel cladding.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 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.

This Specification contains P/T limit curves for heatup, cooldown, and inservice leakage and hydrostatic testing, criticality, and data for the maximum rate of change of reactor coolant temperature.

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 P/T limit curves apply to the reactor pressure vessel, since the vessel is the component most subject to brittle failure, and is bounding over other SSCs that comprise the reactor coolant pressure boundary. The fluid temperatures of an idle recirculation loop are not representative of reactor vessel conditions and therefore the P/T limit curves do not apply to an idle recirculation loop.

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 most subject 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, abnormal operational transients, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 2). The NRC has also approved the use of alternate fracture toughness curves for establishing these limits (Ref. 10).

(continued)

BASES

BACKGROUND
(continued)

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with BWRVIP-86-A (Ref. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 5.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limits include the Reference 1 requirement that they be at least 40°F above the heatup curve or the cooldown curve and at least 60°F above the adjusted reference temperature of the reactor vessel material in the region that is controlling (reactor vessel flange region) (Ref. 6).

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the reactor pressure vessel, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE
SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the reactor pressure vessel, a condition that is unanalyzed. Since the

(continued)

BASES

APPLICABLE P/T limits are not derived from any DBA, there are no acceptance limits
SAFETY ANALYSES related to the P/T limits. Rather, the P/T limits are acceptance limits
(continued) themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 8).

LCO

The elements of this LCO are:

- a. RCS pressure and temperature (Bellline, Bottom Head, and Upper Vessel) are within the applicable limits of Figure 3.4.9-1 and Figure 3.4.9-2, and heatup or cooldown rates are $\leq 100^{\circ}\text{F}$ when averaged over a one hour period during RCS heatup, cooldown, and inservice leak and hydrostatic testing (The Adjusted Reference Temperature (ART) bellline region must be determined from Figure 3.4.9-2;
- b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is $\leq 145^{\circ}\text{F}$ during recirculation pump startup;
- c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is $\leq 50^{\circ}\text{F}$ during recirculation pump startup;
- d. RCS pressure and temperature are within the criticality limits specified in Figure 3.4.9-3, prior to achieving criticality; and
- e. The reactor vessel flange and the head flange temperatures are $> 80^{\circ}\text{F}$ when tensioning the reactor vessel head bolting studs.

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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.

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored.

Besides restoring the P/T limit parameters to within limits, an evaluation is required to determine if RCS operation is allowed. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 212°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation; however, its use is restricted to evaluation of the bellline.

Condition C is modified by a Note requiring Required Action C.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the reactor pressure vessel integrity.

SURVEILLANCE
REQUIREMENTS

SR 3.4.9.1

Verification that operation is within RCS pressure, RCS temperature, and RCS heatup and cooldown rate limits by monitoring the bottom head drain, recirculation loop temperatures, and RPV metal temperatures (Bellline, Bottom Head, and Upper Vessel) is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7 (continued)

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

SR 3.4.9.5 is modified by a Note that requires the Surveillance to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.9.6 is modified by a Note that requires the Surveillance to be initiated 30 minutes after RCS temperature $\leq 90^{\circ}\text{F}$ in MODE 4. SR 3.4.9.7 is modified by a Note that requires the Surveillance to be initiated 12 hours after RCS temperature $\leq 100^{\circ}\text{F}$ in MODE 4. The Notes contained in these SRs are necessary to specify when the reactor vessel flange and head flange temperatures are required to be within the specified limits.

REFERENCES

1. 10 CFR 50, Appendix G.
 2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
 3. BWRVIP-86-A, October 2002.
 4. 10 CFR 50, Appendix H.
 5. Regulatory Guide 1.99, Revision 2, May 1988.
 6. USAR, Section IV-2.6.
 7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
 8. 10 CFR 50.36(c)(2)(ii).
 9. USAR, Appendix G.
 10. ASME XI Code Case N-640.
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BASES

BACKGROUND (continued)

Water from the break returns to the suppression pool where it is used again and again. Water in the suppression pool is circulated through a heat exchanger cooled by the RHR Service Water Booster System. Depending on the location and size of the break, portions of the ECCS may be ineffective; however, the overall design is effective in cooling the core regardless of the size or location of the piping break.

All ECCS subsystems are designed to ensure that no single active component failure will prevent automatic initiation and successful operation of the minimum required ECCS equipment.

The CS System (Ref. 1) is composed of two independent subsystems. Each subsystem consists of a motor driven pump, a spray sparger above the core, and piping and valves to transfer water from the suppression pool to the sparger. The CS System is designed to provide cooling to the reactor core when reactor pressure is low. Upon receipt of an initiation signal, the CS pumps in both subsystems are automatically started, following an approximate 10 second time delay, when AC power is available. When the RPV pressure drops sufficiently, CS System flow to the RPV begins. A full flow test line is provided to route water to the suppression pool to allow testing of the CS System without spraying water in the RPV.

LPCI is an independent operating mode of the RHR System. There are two LPCI subsystems (Ref. 2), each consisting of two motor driven pumps in parallel and piping and valves to transfer water from the suppression pool to the RPV via the corresponding recirculation loop. Two RHR pumps, denoted A and C, discharge into A subsystem, and two RHR pumps, denoted B and D, discharge into B subsystem. Pumps A and B are powered from Division I (4160VAC F), and pumps C and D are powered from Division II (4160VAC G). The two LPCI subsystems can be interconnected via the RHR System cross tie shutoff valve; however, the cross tie shutoff valve is maintained closed to prevent loss of both LPCI subsystems during a LOCA. The LPCI subsystems are designed to provide core cooling at low RPV pressure. Upon receipt of an initiation signal, all four LPCI pumps are automatically started (pumps A and D immediately when AC power is available, and pumps B and C approximately 5 seconds after AC power is available). RHR System valves in the LPCI flow path are automatically positioned to ensure the proper flow path for water from the suppression pool to inject into the recirculation loops. When the RPV pressure drops sufficiently, the LPCI flow to the RPV, via the

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

e. Adequate long term cooling capability is maintained.

The limiting single failures are discussed in Reference 9. For large or small beak LOCA, failure of a DC power source is considered the most severe failure. Credit is taken for 5 of 6 ADS valves. The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage.

The ECCS satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 10).

LCO

Each ECCS injection/spray subsystem and six ADS valves are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the two CS subsystems, the two LPCI subsystems, and one HPCI System. The low pressure ECCS injection/spray subsystems are defined as the two CS subsystems and the two LPCI subsystems.

With less than the required number of ECCS subsystems OPERABLE, the potential exists that during a limiting design basis LOCA concurrent with the worst case single failure, the limits specified in Reference 8 could be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by Reference 8.

Two pumps are required for an OPERABLE LPCI subsystem. However, continued operation is permitted for a period of seven days with one pump inoperable in both subsystems. LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below the actual RHR cut in permissive pressure in MODE 3, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. Alignment and operation for decay heat removal includes when the required RHR pump is not operating or when the system is realigned from or to the RHR shutdown cooling mode. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling when necessary.

(continued)

BASES (continued)

APPLICABILITY All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3, when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system piping. In MODES 2 and 3, when reactor steam dome pressure is ≤ 150 psig, ADS and HPCI are not required to be OPERABLE because the low pressure ECCS subsystems can provide sufficient flow below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2, "ECCS — Shutdown."

ACTIONSA.1

If any one low pressure ECCS injection/spray subsystem is inoperable, or if one LPCI pump in both LPCI subsystems is inoperable, the inoperable subsystem(s) must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, because a single failure in one of the remaining OPERABLE subsystems, concurrent with a LOCA, may result in the ECCS not being able to perform its intended safety function. The 7 day Completion Time is consistent with the recommendations provided in a reliability study (Ref. 11) that evaluated the impact on ECCS availability, assuming various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed Completion Times.

B.1 and B.2

If the inoperable low pressure ECCS subsystem cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 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.

(continued)

BASES

ACTIONS
(continued)C.1 and C.2

If the HPCI System is inoperable and the RCIC System is verified to be OPERABLE, the HPCI System must be restored to OPERABLE status within 14 days. In this condition, adequate core cooling is ensured by the OPERABILITY of the redundant and diverse low pressure ECCS injection/spray subsystems in conjunction with ADS. Also, the RCIC System will automatically provide makeup water at most reactor operating pressures. Verification of RCIC OPERABILITY within 1 hour is therefore required when HPCI is inoperable. This may be performed as an administrative check by examining logs or other information, to determine if RCIC is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of the RCIC System. If the OPERABILITY of the RCIC System cannot be verified, however, Condition G must be immediately entered. If a single active component fails concurrent with a design basis LOCA, there is a potential, depending on the specific failure, that the minimum required ECCS equipment will not be available. A 14 day Completion Time is consistent with the recommendations provided in a reliability study cited in Reference 11 and has been found to be acceptable through operating experience.

D.1 and D.2

If any one low pressure ECCS injection/spray subsystem, or one LPCI pump in both LPCI subsystems, is inoperable in addition to an inoperable HPCI System, the inoperable low pressure ECCS injection/spray subsystem or the HPCI System must be restored to OPERABLE status within 72 hours. In this condition, adequate core cooling is ensured by the OPERABILITY of the ADS and the remaining low pressure ECCS subsystems. However, the overall ECCS reliability is significantly reduced because a single failure in one of the remaining OPERABLE subsystems concurrent with a design basis LOCA may result in the ECCS not being able to perform its intended safety function. Since both a high pressure system (HPCI) and a low pressure subsystem are inoperable, a more restrictive Completion Time

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

of 72 hours is required to restore either the HPCI System or the low pressure ECCS injection/spray subsystem to OPERABLE status. This Completion Time is consistent with the recommendations provided in a reliability study cited in Reference 11 and has been found to be acceptable through operating experience.

E.1

The LCO requires six ADS valves to be OPERABLE in order to provide the ADS function. Reference 9 contains the results of an analysis that evaluated the effect of one ADS valve being out of service. This analysis shows that, assuming a failure of the HPCI System, operation of only five ADS valves will provide the required depressurization. However, overall reliability of the ADS is reduced, because a single failure in the OPERABLE ADS valves could result in a reduction in depressurization capability. Therefore, operation is only allowed for a limited time. The 14 day Completion Time is consistent with the recommendations provided in a reliability study cited in Reference 11 and has been found to be acceptable through operating experience.

F.1 and F.2

If any one low pressure ECCS injection/spray subsystem, or one LPCI pump in both LPCI subsystems, is inoperable in addition to one ADS valve inoperable, adequate core cooling is ensured by the OPERABILITY of HPCI and the remaining low pressure ECCS injection/spray subsystem. However, overall ECCS reliability is reduced because a single active component failure concurrent with a design basis LOCA could result in the minimum required ECCS equipment not being available. Since both a high pressure system (ADS) and a low pressure subsystem are inoperable, a more restrictive Completion Time of 72 hours is required to restore either the low pressure ECCS subsystem or the ADS valve to OPERABLE status. This Completion Time is consistent with the recommendations provided in a reliability study cited in Reference 11 and has been found to be acceptable through operating experience.

(continued)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.2 ECCS — Shutdown

BASES

BACKGROUND A description of the Core Spray (CS) System and the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System is provided in the Bases for LCO 3.5.1, "ECCS — Operating."

APPLICABLE SAFETY ANALYSES The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated loss of coolant accident (LOCA). The long term cooling analysis following a design basis LOCA (Ref. 1) demonstrates that only one low pressure ECCS spray subsystem is required, post LOCA, to provide sufficient heat removal and maintain adequate reactor vessel water level. It is reasonable to assume, based on engineering judgement, that while in MODES 4 and 5, one low pressure ECCS injection/spray subsystem can maintain adequate reactor vessel water level in the event of an inadvertent vessel draindown. To provide redundancy, a minimum of two low pressure ECCS injection/spray subsystems are required to be OPERABLE in MODES 4 and 5.

The low pressure ECCS subsystems satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii) (Ref. 2).

LCO Two low pressure ECCS injection/spray subsystems are required to be OPERABLE. The low pressure ECCS injection/spray subsystems consist of two CS subsystems and two LPCI subsystems. Each CS subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool or condensate storage tank (CST) to the reactor pressure vessel (RPV). Each LPCI subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool or CST to the RPV. Only a single LPCI pump is required per subsystem because of the larger injection capacity in relation to a CS subsystem. In MODES 4 and 5, the RHR System cross tie shutoff valve is not required to be closed. The necessary portions of the

(continued)

BASES

SURVEILLANCE
REQUIREMENTSSR 3.5.2.3 (continued)

operation. This SR applies only to valves affecting the direct flow path. This SR excluded valves that, if mispositioned, would not affect system or subsystem OPERABILITY. Also, this SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is appropriate because the valves are operated under procedural control and the probability of their being mispositioned during this time period is low.

In MODES 4 and 5, the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Therefore, this SR is modified by a Note that allows one LPCI subsystem to be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. Alignment and operation for decay heat removal includes when the required RHR pump is not operating or when the system is being realigned from or to the RHR shutdown cooling mode. Because of the low pressure and low temperature conditions in MODES 4 and 5 sufficient time will be available to manually align and initiate LPCI subsystem operation to provide core cooling prior to postulated fuel uncover. This will ensure adequate core cooling if an inadvertent RPV draindown should occur.

REFERENCES

1. NEDO-20566A, "General Electric Company Analytical Model for Loss-of-Coolant Analysis In Accordance With 10CFR50 Appendix K," September 1986.
 2. 10 CFR 50.36(c)(2)(ii).
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BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by Note 1 that says the Surveillance is not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The time allowed for this test after required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. Adequate reactor pressure must be available to perform this test. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Thus, sufficient time is allowed after adequate pressure and flow are achieved to perform this test. Adequate reactor steam pressure is 145 psig. Adequate steam flow is represented by turbine bypass valves at least 30% open, or a total steam flow of 10^6 lb/hr. Reactor startup is allowed prior to performing this test because the reactor pressure is low and the time allowed to satisfactorily perform the test is short. For SR 3.5.3.3, while adequate pressure can be reached prior to the required Applicability for RCIC, the 12 hour allowance of the Note would not apply until entering the Applicability (>150 psig) with adequate steam flow.

This SR is modified by Note 2 that excludes vessel injection during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

REFERENCES

1. USAR, Appendix F, Section F-2.2.1.
 2. USAR, Section IV-7.
 3. 10 CFR 50.36(c)(2)(ii).
 4. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
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BASES (continued)

**SURVEILLANCE
REQUIREMENTS**

SR 3.6.4.2.1

This SR verifies that each secondary containment manual isolation valve and blind flange that is not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the secondary containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those SCIVs in secondary containment that are capable of being mispositioned are in the correct position.

Since these SCIVs are readily accessible to personnel during normal operation and verification of their position is relatively easy, the 31 day Frequency was chosen to provide added assurance that the SCIVs are in the correct positions. This SR does not apply to valves and blind flanges that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

Two Notes have been added to this SR. The first Note applies to valves and blind flanges located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these isolation devices, once they have been verified to be in the proper position, is low.

A second Note has been included to clarify that SCIVs that are open under administrative controls are not required to meet the SR during the time the SCIVs are open. These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

SR 3.6.4.2.2

Verifying that the isolation time of each power operated automatic SCIV is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the SCIV will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SGT) System

BASES

BACKGROUND The SGT System is described in the USAR, (Ref. 2). The function of the SGT System is to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) and secondary containment isolation are filtered and adsorbed prior to exhausting to the environment.

The SGT System consists of two fully redundant subsystems, each with its own set of ductwork, dampers, charcoal filter train, and controls. Both SGT subsystems share a common inlet plenum. This inlet plenum is connected to the reactor building exhaust plenum, the primary containment, and the HPCI turbine gland seal exhaust. Both SGT subsystems exhaust to the elevated release point (ERP) tower through a common exhaust duct served by two 100% capacity system fans. Both fans automatically start on a secondary containment isolation signal.

The SGT subsystem fan suctions are cross connected by a single duct and a throttled and locked manual cross tie valve to accommodate decay heat removal from the shutdown SGT subsystem. SGT room air enters the train suction through a check valve and air operated damper, is drawn through the filter removing decay heat from the shutdown SGT subsystem, passes through the cross tie ductwork to the operating SGT subsystem fan, and is exhausted to the ERP tower.

Each charcoal filter train consists of (components listed in order of the direction of the air flow):

- a. A demister or moisture separator;
- b. A rough prefilter;
- c. An electric heater;
- d. A high efficiency particulate air (HEPA) filter;
- e. A charcoal adsorber;
- f. A second HEPA filter; and

(continued)

BASES

LCO
(continued)

subsystem consists of a demister, prefilter, HEPA filter, charcoal adsorber, a final HEPA filter, exhaust fan, and associated ductwork, dampers, valves and controls.

When the required decay heat removal flow through the cross tie damper is not met, only ONE SGT subsystem may be considered OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System in OPERABLE status is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status in 7 days. In this Condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT subsystem and the low probability of a DBA occurring during this period.

B.1 and B.2

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1 (continued)

fan motors and controls and the redundancy available in the system.

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

This SR verifies that each SGT subsystem starts on receipt of an actual or simulated initiation signal. While this Surveillance can be performed with the reactor at power, operating experience has shown that these components will pass the Surveillance when performed at the 18 month Frequency. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. Therefore, the Frequency was found to be acceptable from a reliability standpoint.

SR 3.6.4.3.4

This SR verifies that the SGT units cross tie damper is in the correct position, and that each SGT room air supply check valve and each air operated SGT dilution air shutoff valve open when required. This ensures that the decay heat removal function of SGT System operation is available. While this Surveillance can be performed with the reactor at power, operating experience has shown that these components will pass the Surveillance when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was found to be acceptable from a reliability standpoint.

REFERENCES

1. (Deleted)
2. USAR, Section V-3.3.4.
3. 10 CFR 50.36(c)(2)(ii).

BASES

LCO
(continued)

single active failure occurs coincident with the loss of offsite power.

An RHRSWB subsystem is considered OPERABLE when:

- a. Two pumps are OPERABLE; and
- b. An OPERABLE flow path is capable of taking suction from the Service Water System and transferring the water to the required RHR heat exchangers at the assumed flow rate; and
- c. The associated manual valve on the RHRSWB System cross tie piping (which allows the two RHRSWB subsystems to be connected) is closed.

An adequate suction source is not addressed in this LCO since the minimum net positive suction head (45 ft) is bounded by the service water pump requirements (LCO 3.7.2, "Service Water (SW) System and Ultimate Heat Sink (UHS)").

APPLICABILITY

In MODES 1, 2, and 3, the RHRSWB System is required to be OPERABLE to support the OPERABILITY of the RHR System for primary containment cooling (LCO 3.6.2.3, "Residual Heat Removal (RHR) Suppression Pool Cooling") and decay heat removal (LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System — Hot Shutdown"). The Applicability is therefore consistent with the requirements of these systems.

In MODES 4 and 5, the OPERABILITY requirements of the RHRSWB System are determined by the systems it supports and therefore, the requirements are not the same for all facets of operation in MODES 4 and 5. Thus, the LCOs of the RHR Shutdown Cooling System (LCO 3.4.8, "RHR Shutdown Cooling System—Cold Shutdown," LCO 3.9.7, "RHR—High Water Level," and LCO 3.9.8, "RHR—Low Water Level"), which require portions of the RHRSWB System to be OPERABLE, will govern RHRSWB System operation in MODES 4 and 5.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources — Operating

BASES

BACKGROUND The unit AC Sources for the Class 1E AC Electrical Power Distribution System consist of the offsite power sources (preferred power sources, normal and alternates), and the onsite standby power sources (diesel generators (DGs)). As summarized in the USAR, (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The Class 1E AC distribution system is divided into redundant load groups, so loss of any one group does not prevent the minimum safety functions from being performed. Each load group has connections to two qualified offsite power supplies and a single DG.

The offsite power sources are a startup station service transformer (SSST) which connects to the 161 kV switchyard and a separate emergency station service transformer (ESST) energized by a 69 kV line. The 161 kV switchyard is connected to one 161 kV line which terminates in a switchyard near Auburn, Nebraska, and the 345/161 kV, 300 MVA auto-transformer which connects to the 345 kV switchyard. The 345 kV switchyard has five lines which terminate in switchyards near Booneville, Iowa; Hallam, Nebraska; St. Joseph, Missouri; Fairport, Missouri; and Nebraska City, Nebraska. The ESST is fed by a 69 kV line which is part of a subtransmission grid of the Omaha Public Power District. If the normal station service transformer (NSST) (powered by the main generator) is lost, the SSST, which is normally energized, will automatically energize 4160 volt buses 1A and 1B, as well as their connected loads, including critical buses 1F & 1G. If the SSST fails to energize the critical buses, the ESST, which is normally energized, will automatically energize both critical buses. If the ESST were also to fail, the emergency diesel generators would automatically energize their respective buses. A detailed description of the offsite power network and circuits to the onsite Class 1E critical buses is found in the USAR, Sections VIII-2.0 and VIII-3.0 (Ref. 2).

BASES

ACTIONS (continued)

B.1

To ensure a highly reliable power source remains with one DG inoperable, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions must then be entered.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed to be powered from redundant safety related divisions. Redundant required features failures consist of inoperable features associated with a division redundant to the division that has an inoperable DG.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

- a. An inoperable DG exists; and
- b. A redundant required feature on the other division is inoperable.

If, at any time during the existence of this Condition (one DG inoperable), a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

The Control Room Emergency Filter System (CREFS) is a single train system that has required redundant features consisting of a supply fan, and a manual transfer switch for aligning the emergency booster fan and the exhaust booster fan to one energized critical bus capable of being powered from an OPERABLE diesel generator. Compliance with B.2 requires ensuring the redundant supply fan is in service, and manual transfer switch alignment of the other fans to the redundant critical bus within the 4-hour Completion Time.

BASES

ACTIONS B.2 (continued)

Discovering one DG inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE DG results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the station to transients associated with shutdown.

The remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DGs. If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other DG(s), they are declared inoperable upon discovery, and Condition E of LCO 3.8.1 is entered. Once the failure is repaired, and the common cause failure no longer exists, Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of the remaining DG.

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is a reasonable time to confirm that the OPERABLE DG is not affected by the same problem as the inoperable DG.

BASES

ACTIONS B.4
(continued)

In Condition B, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 7 day Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

-----Temporary Note-----

The 7 day completion time to restore a DG to OPERABLE status is temporarily extended to 14 days if the DG is inoperable due to alignment to a fuel oil storage tank that is drained in support of tank cleaning and coating maintenance activities. The inoperable DG must be available to start and load. The day tank level for the inoperable DG must be maintained above the low level alarm setpoint to ensure the DG safety function is maintained while actions to supply fuel from the opposite division fuel oil storage tank are being performed. A DG that is inoperable for any other reason must be restored within the 7 day completion time requirement. This temporary extension of Condition B.4 Completion Time does not apply to the 14 day maximum Completion Time. The maximum time allowed for any combination of required AC power sources to be inoperable remains 14 days. This temporary extension to Condition B.4 Completion Time expires upon completion of the fuel oil storage tank cleaning and coating maintenance activity, but no later than November 30, 2004.

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 7 days. This situation could lead to a total of 14 days, since initial failure of the LCO, to restore the DG. At this time, an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 7 days (for a total of 21 days) allowed prior to complete restoration of the LCO. The 14 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 7 day and 14 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive must be met.

(continued)

BASES

ACTIONS B.4 (continued)
(continued)

Similar to Required Action B.2, the second Completion Time of Required Action B.4 allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This exception results in establishing the "time zero" at the time that the LCO was initially not met, instead of the time that Condition B was entered.

C.1 and C.2

Required Action C.1 addresses actions to be taken in the event of inoperability of redundant required features concurrent with inoperability of two offsite circuits. Required Action C.1 reduces the vulnerability to a loss of function. The Completion Time for taking these actions is reduced to 12 hours from that allowed with one division without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 8) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety divisions are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are designed with redundant safety related divisions, (i.e., single division systems are not included in the list). Redundant required features failures consist of any of these features that are inoperable because any inoperability is on a division redundant to a division with inoperable offsite circuits.

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

- a. Both offsite circuits are inoperable; and
- b. A redundant required feature is inoperable.

If, at any time during the existence of this Condition (both offsite circuits inoperable), a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

According to the recommendations in Regulatory Guide 1.93 (Ref. 8), operation may continue in Condition C for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system may not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have

(continued)

BASES

ACTIONS E.1 (continued)

however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation. According to the recommendations in Regulatory Guide 1.93 (Ref. 8), with both DGs inoperable, operation may continue for a period that should not exceed 2 hours.

F.1 and F.2

If the inoperable AC electrical power sources cannot be restored to OPERABLE status within the associated Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to MODE 4 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.

G.1

Condition G corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The station is required by LCO 3.0.3 to commence a controlled shutdown.

SURVEILLANCE
REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function. Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are in general conformance with the recommendations of Regulatory Guide 1.9 (Ref. 9), Regulatory Guide 1.108 (Ref. 10), and Regulatory Guide 1.137 (Ref. 11).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.11 (continued)

testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 18 months.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil being periodically circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

REFERENCES

1. USAR, Section VIII-1.0.
2. USAR, Section VIII-2.0 and VIII-3.0.
3. Safety Guide 9, Revision 0, March 1971.
4. USAR, Chapter VI.
5. USAR, Chapter XIV.
6. 10 CFR 50.36(c)(2)(ii).
7. Generic Letter 84-15.
8. Regulatory Guide 1.93.
9. Regulatory Guide 1.9, Revision 3, July 1993.
10. Regulatory Guide 1.108.
11. Regulatory Guide 1.137.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES The initial conditions of Design Basis Accident (DBA) and transient analyses in USAR, Chapter VI (Ref. 4), and Chapter XIV (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6, Containment Systems.

Since diesel fuel oil, lube oil, and starting air subsystems support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 6).

-----Temporary Note-----

The Limiting Condition for Operation is modified by a temporary note indicating temporary storage tanks may be used on a one time basis during tank cleaning and coating maintenance activities. Fuel stored in the temporary tanks, in conjunction with fuel oil in one permanent storage tank, may be utilized to maintain the DG aligned to the permanent storage tank OPERABLE. A temporary transfer pump with a capacity greater than 5 gpm must be pre-staged and available to transfer the off-loaded fuel to the DG. This is considered sufficient based on fuel in the permanent tank providing a minimum 4 days full load operation of the DG, contingency measures which pre-stage equipment necessary to transfer the fuel in the temporary tanks to the permanent tank or directly to the DG day tank, and the initiation of actions to obtain replenishment fuel. This temporary note expires upon completion of the fuel oil storage tank cleaning and coating maintenance activity, but no later than November 30, 2004.

LCO

Stored diesel fuel oil is required in sufficient supply for 7 days of operation at maximum post-LOCA load demand. It is also required to meet specific standards for quality. Additionally, sufficient lube oil supply

(continued)

BASES (continued)

LCO
(continued)

must be available to ensure the capability to operate for 7 days at maximum post-LOCA load demand. This requirement, in conjunction with an ability to obtain replacement supplies within 7 days, supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an abnormal operational transient or a postulated DBA with loss of offsite power. DG day tank fuel oil requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources — Operating," and LCO 3.8.2, "AC Sources — Shutdown."

The starting air system is required to have a minimum capacity for multiple DG start attempts in accordance with Reference 7, without recharging the air start receivers. Only one air receiver (and associated airstart header) per DG is required, since each air receiver has the required capacity.

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an abnormal operational transient or a postulated DBA. Because stored diesel fuel oil, lube oil, and starting air subsystems support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil, lube oil, and starting air are required to be within limits when the associated DG is required to be OPERABLE.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG except for Conditions A, C, and D. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) governed by separate Condition entry and application of associated Required Actions. The Note does not apply to Conditions A, C and D since the CNS design has two fuel oil storage tanks that supply fuel oil to both DGs.

(continued)

BASES (continued)

Actions
(continued)

A.1

With the combined fuel level < 49,500 gallons in the storage tanks, the 7 day fuel oil supply for both DGs is not available. The 49,500 gallon limit is a conservative estimate of the required fuel oil based on worst case fuel consumption. However, the Condition is restricted to fuel oil level reductions that maintain at least a 6 day supply (42,800 gallons). These circumstances may be caused by events such as:

- a. Full load operation required for an inadvertent start while at minimum required level; or
- b. Feed and bleed operations that may be necessitated by increasing particulate levels or any number of other oil quality degradations.

This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of the fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DGs inoperable. This period is acceptable based on the remaining capacity (> 6 days), the fact that action will be initiated to obtain replenishment, and the low probability of an event during this brief period.

B.1

With lube oil inventory < 504 gal, sufficient lube oil to support 7 days of continuous DG operation at full load conditions may not be available. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6 day supply. This restriction allows sufficient time for obtaining the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that action will be initiated to obtain replenishment, and the low probability of an event during this brief period.

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion for particulates. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of

(continued)

BASES (continued)

Actions
(continued)

C.1 (continued)

particulates does not mean failure of the fuel oil to burn properly in the diesel engine, since particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and since proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the DGs inoperable. The 7 day Completion Time allows for further evaluation, resampling, and re-analysis of the DG fuel oil.

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.3 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combination of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is high likelihood that the DG would still be capable of performing its intended function. If the new fuel has not yet been added to the fuel oil storage tanks, entry into this Condition is not necessary.

E.1

With starting air receiver pressure < 200 psig, sufficient capacity for multiple DG start attempts in accordance with Reference 7 may not exist. However, as long as the receiver pressure is > 125 psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

F.1

With a Required Action and associated Completion Time of Condition A, B, C, D, or E not met, or the stored diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than addressed by Conditions A, B, C, D, or E, the associated DG(s) may be incapable of performing its intended function and must be immediately declared inoperable.

(continued)

BASES

ACTIONS

A.1 (continued)

remaining division by stabilizing the unit, and on restoring power to the affected division. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because of:

- a. The potential for decreased safety if the unit operators' attention is diverted from the evaluations and actions necessary to restore power to the affected division to the actions associated with taking the unit to shutdown within this time limit.
- b. The potential for an event in conjunction with a single failure of a redundant component in the division with AC power. (The redundant component is verified OPERABLE in accordance with Specification 5.5.11, "Safety Function Determination Program (SFDP).")

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DC bus is inoperable and subsequently returned OPERABLE, this LCO may already have been not met for up to 2 hours. This situation could lead to a total duration of 10 hours, since initial failure of the LCO, to restore the AC electrical power distribution system. At this time a DC bus could again become inoperable, and the AC electrical power distribution system could be restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This results in establishing the "time zero" at the time this LCO was initially not met, instead of at the time Condition A was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

(continued)

BASES

ACTIONS

B.1 (continued)

The 2 hour Completion Time for 125 V DC electrical power distribution subsystems is consistent with Regulatory Guide 1.93 (Ref. 3).

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an AC bus is inoperable and subsequently restored OPERABLE, this LCO may already have been not met for up to 8 hours. This situation could lead to a total duration of 10 hours, since initial failure of the LCO, to restore the DC electrical power distribution system. At this time, an AC bus could again become inoperable, and DC electrical power distribution could be restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This allowance results in establishing the "time zero" at the time this LCO was initially not met, instead of at the time Condition B was entered. The 16 hour Completion Time is an acceptable limitation on this potential of failing to meet the LCO indefinitely.

C.1 and C.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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B 3.9 REFUELING OPERATIONS

B 3.9.1 Refueling Equipment Interlocks

BASES

BACKGROUND Refueling equipment interlocks restrict the operation of the refueling equipment or the withdrawal of control rods to reinforce unit procedures that prevent the reactor from achieving criticality during refueling. The refueling interlock circuitry senses the conditions of the refueling equipment and the control rods. Depending on the sensed conditions, interlocks are actuated to prevent the operation of the refueling equipment or the withdrawal of control rods.

The USAR, Appendix F, specifies that at least one of the two required independent reactivity control systems provided be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies (Ref. 1). The control rods, when fully inserted, serve as the system capable of maintaining the reactor subcritical in cold conditions during all fuel movement activities and accidents.

One channel of instrumentation is provided to sense the position of the refueling platform, the loading of the refueling platform fuel grapple, and the full insertion of all control rods. Additionally, inputs are provided for the loading of the refueling platform frame mounted hoist, the loading of the refueling platform monorail mounted hoist, the not full up position of the fuel grapple, and the loading of the service platform hoist. With the reactor mode switch in the shutdown or refueling position, the indicated conditions are combined in logic circuits to determine if all restrictions on refueling equipment operations and control rod insertion are satisfied.

A control rod not at its full-in position with the grapple not full up interrupts power to the refueling equipment to prevent operating the equipment over the reactor core when loaded with a fuel assembly. Conversely, with the refueling equipment loaded with fuel or the grapple not full up, and located near or over the core, a control rod withdrawal block is inserted in the Reactor Manual Control System to prevent withdrawing a control rod.

(continued)

BASES

ACTIONS

A.1 (continued)

immediately suspended. This action ensures that operations are not performed with equipment that would potentially not be blocked from unacceptable operations (e.g., loading fuel into a cell with a control rod withdrawn). Suspension of in-vessel fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

Performance of a CHANNEL FUNCTIONAL TEST demonstrates each required refueling equipment interlock will function properly when a simulated or actual signal indicative of a required condition is injected into the logic. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

The 7 day Frequency is based on engineering judgment and is considered adequate in view of other indications of refueling interlocks and their associated input status that are available to unit operations personnel.

REFERENCES

1. USAR, Appendix F, Section F-2.5.
 2. USAR, Section VII-6.
 3. USAR, Section XIV-5.3.3.
 4. USAR, Section XIV-5.3.4.
 5. 10 CFR 50.36(c)(2)(ii).
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B 3.9 REFUELING OPERATIONS

B 3.9.2 Refuel Position One-Rod-Out Interlock

BASES

BACKGROUND

The refuel position one-rod-out interlock restricts the movement of control rods to reinforce unit procedures that prevent the reactor from becoming critical during refueling operations. During refueling operations, no more than one control rod is permitted to be withdrawn.

The USAR, Appendix F, specifies that at least one of the two required independent reactivity control systems provided be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies (Ref. 1). The control rods serve as the system capable of maintaining the reactor subcritical in cold conditions.

The refuel position one-rod-out interlock prevents the selection of a second control rod for movement when any other control rod is not fully inserted (Ref. 2). It is a logic circuit that has redundant channels. It uses the all-rods-in signal (from the control rod full-in position indicators discussed in LCO 3.9.4, "Control Rod Position Indication") and a rod selection signal (from the Reactor Manual Control System).

This Specification ensures that the performance of the refuel position one-rod-out interlock in the event of a Design Basis Accident meets the assumptions used in the safety analysis of Reference 3.

APPLICABLE SAFETY ANALYSES

The refueling position one-rod-out interlock is explicitly assumed in the USAR analysis for 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.

The refuel position one-rod-out interlock and adequate SDM(LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" prevent criticality by preventing withdrawal of more than one control rod. With one control rod withdrawn, the core will remain subcritical, thereby preventing any prompt critical excursion.

The refuel position one-rod-out interlock satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

BASES

- REFERENCES
1. USAR, Appendix F, Section F-2.5.
 2. USAR, Section VII-6.
 3. USAR, Section XIV-5.3.3.
 4. 10 CFR 50.36(c)(2)(ii).
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B 3.9 REFUELING OPERATIONS

B 3.9.3 Control Rod Position

BASES

BACKGROUND Control rods provide the capability to maintain the reactor subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the Reactor Manual Control System. During refueling, movement of control rods is limited by the refueling interlocks (LCO 3.9.1, "Refueling Equipment Interlocks," and LCO 3.9.2, "Refuel Position One-Rod-Out Interlock") or the control rod block with the reactor mode switch in the shutdown position (LCO 3.3.2.1, "Control Rod Block Instrumentation").

The CNS USAR, Appendix F, specifies that at least one of the two required independent reactivity control systems provided be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies (Ref. 1). The control rods serve as the system capable of maintaining the reactor subcritical in cold conditions.

The refueling interlocks allow a single control rod to be withdrawn at any time unless fuel is being loaded into the core. To preclude loading fuel assemblies into the core with a control rod withdrawn, all control rods must be fully inserted (Ref. 2). This prevents the reactor from achieving criticality during refueling operations.

APPLICABLE SAFETY ANALYSES

Prevention and mitigation of prompt reactivity excursions during refueling are provided by the refueling interlocks (LCO 3.9.1 and LCO 3.9.2), the SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"), the intermediate range monitor neutron flux scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), and the control rod block instrumentation (LCO 3.3.2.1).

The safety analysis for the control rod removal error during refueling in the USAR (Ref. 3) assumes the functioning of the refueling interlocks and adequate SDM. The analysis for the fuel assembly insertion error (Ref. 4) assumes all control rods are fully inserted. Thus, prior to fuel reload, all control rods must be fully inserted to minimize

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1 (continued)

The 12 hour Frequency takes into consideration the procedural controls on control rod movement during refueling as well as the redundant functions of the refueling interlocks.

REFERENCES

1. USAR, Appendix F, Section F-2.5.
 2. USAR, Section VII-6.
 3. USAR, Section XIV-5.3.3.
 4. USAR, Section XIV-5.3.4.
 5. 10 CFR 50.36(c)(2)(ii).
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B 3.9 REFUELING OPERATIONS

B 3.9.4 Control Rod Position Indication

BASES

BACKGROUND The full-in position indication channel for each control rod provides necessary information to the refueling interlocks to prevent inadvertent criticalities during refueling operations. During refueling, the refueling interlocks (LCO 3.9.1, "Refueling Equipment Interlocks," and LCO 3.9.2, "Refuel Position One-Rod-Out Interlock") use the full-in position indication channel to limit the operation of the refueling equipment and the movement of the control rods. The absence of the full-in position channel signal for any control rod removes the all-rods-in permissive for the refueling equipment interlocks and prevents fuel loading. Also, this condition causes the refuel position one-rod-out interlock to not allow the withdrawal of any other control rod.

The CNS USAR, Appendix F, specifies that at least one of the two required independent reactivity control systems provided be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies (Ref. 1). The control rods serve as the system capable of maintaining the reactor subcritical in cold conditions.

APPLICABLE SAFETY ANALYSES

Prevention and mitigation of prompt reactivity excursions during refueling are provided by the refueling interlocks (LCO 3.9.1 and LCO 3.9.2), the SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"), the intermediate range monitor neutron flux scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), and the control rod block instrumentation (LCO 3.3.2.1, "Control Rod Block Instrumentation").

The safety analysis for the control rod removal error during refueling (Ref. 2) assumes the functioning of the refueling interlocks and adequate SDM. The analysis for the fuel assembly insertion error (Ref. 3) assumes all control rods are fully inserted. The full-in position indication channel is required to be OPERABLE so that the refueling interlocks can ensure that fuel cannot be loaded with any control rod withdrawn and that no more than one control rod can be withdrawn at a time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 (continued)

that when a control rod is withdrawn, the full-in position indication is not present. The full-in position indication channel is considered inoperable even with the control rod fully inserted, if it would continue to indicate full-in with the control rod withdrawn. Performing the SR each time a control rod is withdrawn is considered adequate because of the procedural controls on control rod withdrawals and the visual indications and alarms available in the control room to alert the operator to control rods not fully inserted.

REFERENCES

1. USAR, Appendix F, Section F-2.5.
 2. USAR, Section XIV-5.3.3.
 3. USAR, Section XIV-5.3.4.
 4. 10 CFR 50.36(c)(2)(ii).
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B 3.9 REFUELING OPERATIONS

B 3.9.5 Control Rod OPERABILITY — Refueling

BASES

BACKGROUND Control rods are components of the Control Rod Drive (CRD) System, the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes during refueling operation. In addition, the control rods provide the capability to maintain the reactor subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System.

The CNS USAR, Appendix F, specifies that at least one of the two required independent reactivity control systems provided be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies (Ref. 1). The CRD System is the system capable of maintaining the reactor subcritical in cold conditions.

APPLICABLE SAFETY ANALYSES

Prevention and mitigation of prompt reactivity excursions during refueling are provided by refueling interlocks (LCO 3.9.1, "Refueling Equipment Interlocks," and LCO 3.9.2, "Refuel Position One-Rod-Out Interlock"), the SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"), the intermediate range monitor neutron flux scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), and the control rod block instrumentation (LCO 3.3.2.1, "Control Rod Block Instrumentation").

The safety analyses for the control rod removal error during refueling (Ref. 2) and the fuel assembly insertion error (Ref. 3) evaluate the consequences of control rod withdrawal during refueling and also fuel assembly insertion with a control rod withdrawn. A prompt reactivity excursion during refueling could potentially result in fuel failure with subsequent release of radioactive material to the environment. Control rod scram provides protection should a prompt reactivity excursion occur.

Control rod OPERABILITY during refueling satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1 and SR 3.9.5.2 (continued)

automatic insertion and the associated CRD scram accumulator pressure is \geq 940 psig.

The 7 day Frequency takes into consideration equipment reliability, procedural controls over the scram accumulators, and control room alarms and indicating lights that indicate low accumulator charge pressures.

SR 3.9.5.1 is modified by a Note that allows 7 days after withdrawal of the control rod to perform the Surveillance. This acknowledges that the control rod must first be withdrawn before performance of the Surveillance, and therefore avoids potential conflicts with SR 3.0.3 and SR 3.0.4.

REFERENCES

1. USAR, Appendix F, Section F-2.5.
 2. USAR, Section XIV-5.3.3.
 3. USAR, Section XIV-5.3.4.
 4. 10 CFR 50.36(c)(2)(ii).
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