Nebraska Public Power District

"Always there when you need us"

NLS2010029
April 15, 2010

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

Dear Sir or Madam:

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 September 5, 2008, through March 4, 2010. 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) 825-2904.

Sincerely,

David W. Van Der Kamp
Licensing Manager

/lb

Enclosure

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

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

Senior Resident Inspector, w/enclosure
USNRC - CNS

NPG Distribution, w/o enclosure

CNS Records, w/enclosure

A001
NRR

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ATTACHMENT 3 LIST OF REGULATORY COMMITMENTS©[4]

Correspondence Number: NLS2010029

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
None		

NLS2010029
ENCLOSURE

TECHNICAL SPECIFICATION
BASES CHANGES

FILING INSTRUCTIONS

TECHNICAL SPECIFICATION BASES

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B 3.0-5	B 3.0-5
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B 3.0-8	B 3.0-8
B 3.0-9	B 3.0-9
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B 3.0-13	B 3.0-13
B 3.0-14	B 3.0-14
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FILING INSTRUCTIONS

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B 3.1-51	B 3.1-51
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B 3.3-76	B 3.3-76
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B 3.3-148	B 3.3-148
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B 3.5-26	B 3.5-26
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B 3.6-7	B 3.6-7
B 3.6-13	B 3.6-13
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B 3.7-8	B 3.7-8
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FILING INSTRUCTIONS

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B 2.0-5	09/25/09	B 3.1-21	12/18/03
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B 2.0-8	09/25/09	B 3.1-24	0
		B 3.1-25	05/09/06
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B 3.0-2	0	B 3.1-27	05/09/06
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B 3.0-5	09/18/09	B 3.1-30	0
B 3.0-6	09/18/09	B 3.1-31	0
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B 3.0-8	09/18/09	B 3.1-33	01/30/03
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B 3.3-22	6/28/01	B 3.3-66	1
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B 3.3-29	02/20/07	B 3.3-73	1
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B 3.3-179	0	B 3.4-12	1
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B 3.3-187	12/22/05	B 3.4-20	0
B 3.3-188	12/22/05	B 3.4-21	0
B 3.3-189	10/05/06	B 3.4-22	0
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B 3.3-192	10/05/06	B 3.4-25	0
B 3.3-193	02/20/07	B 3.4-26	09/18/09
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B 3.3-196	11/04/01	B 3.4-29	09/25/09
B 3.3-197	11/04/01	B 3.4-30	0
B 3.3-198	11/04/01	B 3.4-31	09/18/09
B 3.3-199	11/04/01	B 3.4-32	09/25/09
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B 3.3-201	11/04/01	B 3.4-34	0
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B 3.3-207	11/04/01	B 3.4-40	0
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B 3.4-54	0	B 3.6-11	0
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B 3.5-3	0	B 3.6-16	1
B 3.5-4	0	B 3.6-17	0
B 3.5-5	04/26/04	B 3.6-18	0
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B 3.5-20	0	B 3.6-33	12/27/02
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B 3.5-25	1	B 3.6-38	02/20/07
B 3.5-26	09/18/09	B 3.6-39	0
B 3.5-27	09/18/09	B 3.6-40	0
B 3.5-28	09/18/09	B 3.6-41	0
B 3.5-29	02/20/07	B 3.6-42	0
B 3.5-30	12/18/03	B 3.6-43	0
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B 3.6-54	0	B 3.7-10	05/06/09
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B 3.7-1	1	B 3.8-7	09/18/09
B 3.7-2	0	B 3.8-8	09/18/09
B 3.7-3	03/24/04	B 3.8-9	1/17/05
B 3.7-4	0	B 3.8-10	07/01/04

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B 3.8-12	12/22/05	B 3.8-56	0
B 3.8-13	1	B 3.8-57	0
B 3.8-14	1	B 3.8-58	0
B 3.8-15	12/18/03	B 3.8-59	0
B 3.8-16	1	B 3.8-60	0
B 3.8-17	0	B 3.8-61	0
B 3.8-18	0	B 3.8-62	0
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B 3.8-20	0	B 3.8-64	0
B 3.8-21	0	B 3.8-65	0
B 3.8-22	0	B 3.8-66	0
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B 3.8-38	0	B 3.9-6	05/09/06
B 3.8-39	0	B 3.9-7	05/09/06
B 3.8-40	1	B 3.9-8	05/09/06
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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) for the Control Rod Drop and Main Steam Line Break accidents, and 10 CFR 50.67, "Accident Source Term," limits (Ref. 4) for the Fuel Handling and Loss-of-Coolant accidents. 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.
4. 10 CFR 50.67.

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 50.67, "Accident Source Term" (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

BASES

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 50.67, "Accident Source Term," 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 50.67.
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.

BASES

LCO 3.0.3 (continued)

assemblies in the spent fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.6 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.6 to "Suspend movement of irradiated fuel assemblies in the spent fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

BASES

LCO 3.0.4 (continued)

The risk assessment may used quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented.

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as

BASES

LCO 3.0.4 (continued)

long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

LCO 3.0.4.c allows entry into a MODE of other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states that LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., RCS Specific Activity) and may be applied to other Specifications based on NRC plant-specific approval.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, and MODE 3 to MODE 4.

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

BASES

LCO 3.0.4 (continued)

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.

BASES

LCO 3.0.5 (continued)

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR on another channel in the same trip system.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the plant is maintained in a safe condition are specified in the support systems' LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCO Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable

BASES

LCO 3.0.6 (continued)

Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.11, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross division checks to identify a loss of safety function for those support systems that support safety systems are required. The cross division check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Special Operations LCOs in Section 3.10 allow specified TS requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Special Operations LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Special Operations LCOs is optional. A special operation may be performed either under the provisions of the appropriate Special Operations LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Special Operations LCO, the requirements of the

BASES

LCO 3.0.7 (continued)

Special Operations LCO shall be followed. When a Special Operations LCO requires another LCO to be met, only the requirements of the LCO statement are required to be met regardless of that LCO's Applicability (i.e., should the requirements of this other LCO not be met, the ACTIONS of the Special Operations LCO apply, not the ACTIONS of the other LCO). However, there are instances where the Special Operations LCO ACTIONS may direct the other LCO ACTIONS be met. The Surveillances of the other LCO are not required to be met, unless specified in the Special Operations LCO. If conditions exist such that the Applicability of any other LCO is met, all the other LCO's requirements (ACTIONS and SRs) are required to be met concurrent with the requirements of the Special Operations LCO.

LCO 3.0.8

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

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

LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a can be used only if one of the following two means of heat removal is available:

BASES

LCO 3.0.8 (continued)

(1) At least one high pressure makeup path (e.g., using HPCI or RCIC or equivalent) and heat removal capability including a minimum set of supporting equipment required for success, not associated with the inoperable snubber(s),

OR

(2) At least one low pressure makeup path (e.g., LPCI) and heat removal capability including a minimum set of supporting equipment required for success, not associated with the inoperable snubber(s),

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

LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b can be used only if at least one success path exists, using equipment not associated with the inoperable snubber(s), to provide makeup and core cooling needed to mitigate LOOP accident sequences.

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

LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. Ths risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.10 and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a Special Operations LCO are only applicable when the Special Operations LCO is used as an allowable exception to the requirements of a Specification.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed. Some examples of this process are:

a. Control Rod Drive maintenance during refueling that requires scram testing at > 800 psi. However, if other appropriate testing is satisfactorily completed and the scram time testing of SR 3.1.4.3 is satisfied, the control rod can be considered OPERABLE. This allows startup to proceed to reach 800 psi to perform other necessary testing.

b. High pressure coolant injection (HPCI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPCI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

BASES

SR 3.0.2 (continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 3.0.2 does not apply is in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of the regulations. The TS cannot in and of themselves extend a test interval specified in the regulations.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been

BASES

SR 3.0.3 (continued)

performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met. This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not to have been performed when specified, SR 3.0.3 allows the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, 'Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.'

BASES

SR 3.0.3 (continued)

This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability. However, a provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to an SR not being met if the entry is made in accordance with LCO 3.0.4.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

BASES

SR 3.0.4 (continued)

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed per SR 3.0.1, which states that Surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency, on equipment that is inoperable, does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, and MODE 3 to MODE 4.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

BASES

ACTIONS (continued)

The description of "slow" control rods is provided in LCO 3.1.4, "Control Rod Scram Times." In addition, the associated control rod drive must be disarmed (hydraulically) in 2 hours. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable time to perform the Required Action in an orderly manner. The control rod must be isolated from both scram and normal insert and withdraw pressure. Isolating the control rod from scram and normal insert and withdraw pressure prevents damage to the CRDM. The control rod should be isolated from scram and normal insert and withdraw pressure, while maintaining cooling water to the CRD.

Monitoring of the insertion capability of each withdrawn control rod must also be performed within 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM. SR 3.1.3.3 performs periodic tests of the control rod insertion capability of withdrawn control rods. Testing each withdrawn control rod ensures that a generic problem does not exist. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock". The Required Action A.3 Completion Time only begins upon discovery of Condition A concurrent with THERMAL POWER greater than the actual LPSP of the RWM since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The allowed Completion Time provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests.

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

BASES

ACTIONS (continued)

active function (i.e., scram) of the control rods. The number of control rods permitted to be inoperable when operating above 9.85 RTP (e.g., no CRDA considerations) could be more than the value specified, but the occurrence of a large number of inoperable control rods could be indicative of a generic problem, and investigation and resolution of the potential problem should be undertaken. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The position of each control rod must be determined to ensure adequate information on control rod position is available to the operator for determining control rod OPERABILITY and controlling rod patterns. Control rod position may be determined by the use of OPERABLE position indicators, by moving control rods to a position with an OPERABLE indicator, or by the use of other appropriate methods. The 24 hour Frequency of this SR is based on operating experience related to expected changes in control rod position and the availability of control rod position indications in the control room.

SR 3.1.3.2 (Deleted)

SR 3.1.3.3

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal. This Surveillance is not required when THERMAL POWER is less than or equal to the actual LPSP of the RWM, since the notch insertions may not be compatible with the requirements of the Banked Position Withdrawal Sequence (BPWS) (LCO 3.1.6) and the RWM (LCO 3.3.2.1). Withdrawn control

BASES

SURVEILLANCE REQUIREMENTS (continued)

rods are tested at a 31 day Frequency, based on the potential power reduction required to allow the control rod movement. Furthermore, the 31 day Frequency takes into account operating experience related to changes in CRD performance. At any time, if a control rod is immovable, a determination of that control rod's capability of insertion by scram (OPERABILITY) must be made and appropriate action taken.

This SR is modified by a Note that allows 31 days after withdrawal of the control rod and increasing power to above the LPSP, to perform the Surveillance. This acknowledges that the control rod must be first withdrawn and THERMAL POWER must be increased to above the LPSP before performance of the Surveillance, and therefore the Notes avoid potential conflicts with SR 3.0.3 and SR 3.0.4.

SR 3.1.3.4

Verifying that the scram time for each control rod to notch position 06 is $\leq$ 7 seconds provides reasonable assurance that the control rod will insert when required during a DBA or transient, thereby completing its shutdown function. This SR is performed in conjunction with the control rod scram time testing of SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," and the functional testing of SDV vent and drain valves in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlap this Surveillance to provide complete testing of the assumed safety function. The associated Frequencies are acceptable, considering the more frequent testing performed to demonstrate other aspects of control rod OPERABILITY and operating experience, which shows scram times do not significantly change over an operating cycle.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram.

The SLC System is also used to maintain the pH of the water in the Suppression Chamber at or above 7.0 following a design basis Loss-of-Coolant Accident (LOCA) with indication of fuel damage. Maintaining the pH of the water above 7.0 following a LOCA ensures that iodine will be retained in the Suppression Chamber water as assumed in the LOCA Alternative Source Term analysis (Ref. 4). The SLC System is manually initiated from the main control room upon detection of symptoms that a LOCA with fuel damage is occurring.

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject, using both SLC pumps, a quantity of boron, which

BASES

APPLICABLE SAFETY ANALYSES (continued)

produces a concentration of 660 ppm of natural boron, in the reactor coolant at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an amount of boron equal to 25% of the amount cited above is added (Ref. 2). The volume versus concentration limits in Figure 3.1.7-1 and the temperature versus concentration limits in Figure 3.1.7-2 are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

The Alternative Source Term LOCA analysis methodology (Ref. 4) credits the use of the SLC System for injecting sodium pentaborate solution into the RPV following a LOCA with fuel damage to maintain the pH of the water in the Suppression Chamber above 7.0. By maintaining the pH of the water above 7.0 following a LOCA with fuel damage, the majority of the iodine released from a damaged core will be retained in solution in the water and not released as elemental iodine in gaseous form. This will ensure that the radiological consequences from the LOCA will remain within the limits of 10 CFR 50.67 (Ref. 5). Credit for the SLC System in the radiological analyses is based on operation of one SLC pump, initiated 6 hours after start of the LOCA, with injection completed within 8 hours. (Ref. 2)

The SLC System satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii) (Ref. 3).

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. Additionally, an OPERABLE SLC System allows the injection of the sodium pentaborate solution into the RPV following a LOCA with core damage in order to maintain the pH of the water in the Suppression Chamber at or above 7.0. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

BASES

APPLICABILITY

The SLC System must be OPERABLE in Modes 1, 2, and 3.

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn. In MODES 1, 2, and 3, the SLC System must be OPERABLE in order to inject the boron solution into the RPV following a LOCA with fuel damage for purposes of chemistry control of the water, i.e., to maintain the pH of the combined RCS and ECCS water inventory at or above 7.0. Maintaining the pH of this water at or above 7.0 ensures that the radiological consequences of a LOCA remain within the limits of 10 CFR 50.67. (Ref. 5)

ACTIONS

A.1

If one SLC subsystem is in operable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the original licensing basis shutdown function. However, the overall capability is reduced since the remaining OPERABLE subsystem cannot meet the requirements of Reference 1. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the original licensing basis SLC System function and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive (CRD) System to shut down the plant.

B.1

If both SLC subsystems are inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor.

BASES

ACTIONS (continued)

C.1 and C.2

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution volume and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The temperature versus concentration curve of Figure 3.1.7-2 ensures that a 10°F margin will be maintained above the saturation temperature. The 24 hour Frequency is based on operating experience and has shown there are relatively slow variations in the measured parameters of volume and temperature.

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.6 verifies that each valve in the system is in its correct position, but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for manual valves in the SLC System flow path provides assurance that the proper flow paths will exist for system operation. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This Surveillance also does not apply to valves that are locked, sealed, or otherwise secured in position since they are verified to be in the correct position prior to locking, sealing, or securing. This verification of valve alignment does not require any testing or valve manipulation; rather, it involves verification that those valves capable of 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 based on engineering judgment and is consistent with the procedural controls governing valve operation that ensures correct valve positions.

SR 3.1.7.5

This Surveillance requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron in the storage tank is maintained per Figure 3.1.7-1. SR 3.1.7.5 must be performed anytime boron or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. SR 3.1.7.5 must also be performed anytime the temperature is restored to within the limits of Figure 3.1.7-2, to ensure that no significant boron precipitation occurred. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.7

Demonstrating that each SLC System pump develops a flow rate $\geq$ 38.2 gpm at a discharge pressure $\geq$ 1300 psig, by recirculating demineralized water to the test tank, ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the

BASES

SURVEILLANCE REQUIREMENTS (continued)

positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

SR 3.1.7.8 and SR 3.1.7.9

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 36 months at alternating 18 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. 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; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all heat traced piping between the boron solution storage tank and the suction inlet to the injection pumps is unblocked ensures that there is a functioning flow path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping is unblocked is to manually initiate the system, except the explosive valves, and pump from the storage tank to the test tank. Upon completion of this verification, the pump suction piping must be flushed with demineralized water to ensure piping between the storage tank and pump suction is unblocked. The 18 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. This is especially true in light of the temperature verification of this piping

BASES

SURVEILLANCE REQUIREMENTS (continued)

required by SR 3.1.7.3. However, if, in performing SR 3.1.7.3, it is determined that the temperature of this piping has fallen below the specified minimum, SR 3.1.7.9 must be performed once within 24 hours after the piping temperature is restored to within the limits of Figure 3.1.7-2.

REFERENCES

1. 10 CFR 50.62.
2. USAR, Section III-9.
3. 10 CFR 50.36(c)(2)(ii).
4. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants, February 1995."
5. 10 CFR 50.67, "Accident Source Term."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to separate drain lines with two valves in series for a total of four drain valves. Each header is connected to a separate vent line with two valves in series for a total of four vent valves. The automatic SDV vent and drain valves are air to open, spring closing valves which fail closed on loss of air. The SDV drain valves are equipped with a manual override which can only be used to open the valve. The SDV vent valves do not have this feature. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100 (Ref. 2) for the Control Rod Drop and Main Steam Line Break accidents, and 10 CFR 50.67, "Accident Source Term," (Ref. 5) for the Fuel Handling and Loss-of-Coolant accidents; and

b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are within the limits of 10 CFR 100 (Ref. 2) for the Control Rod Drop and Main Steam Line Break accidents, and 10 CFR 50.67, "Accident Source Term," (Ref. 5) for the Fuel Handling and Loss-of-Coolant accidents, and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level in the instrument volume exceeds a specified setpoint. The setpoint is chosen so that all control rods are inserted before the SDV has insufficient volume to accept a full scram.

SDV vent and drain valves satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii) (Ref. 4).

LCO

The OPERABILITY of all SDV vent and drain valves ensures that the SDV vent and drain valves will close during a scram to contain reactor water discharged to the SDV piping. Since the vent and drain lines are provided with two valves in series, the single failure of one valve in the open position will not impair the isolation function of the system. Additionally, the valves are required to open on scram reset to ensure that a path is available for the SDV piping to drain freely at other times.

APPLICABILITY

In MODES 1 and 2, scram may be required; therefore, the SDV vent and drain valves must be OPERABLE. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. Also, during MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Therefore, the SDV vent and drain valves are not required to be OPERABLE in these MODES since the reactor is subcritical and only one rod may be withdrawn and subject to scram.

BASES

REFERENCES

1. USAR, Section III-5.

2. 10 CFR 100.

3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.

4. 10 CFR 50.36(c)(2)(ii).

5. 10 CFR 50.67, "Accident Source Term."

BASES

APPLICABILITY (continued)

plant conditions are such that the likelihood of an event that would require PAM instrumentation is extremely low; therefore, PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

A Note has been provided to modify the ACTIONS related to PAM instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable PAM instrumentation channels provide appropriate compensatory measures for separate Functions and for separate penetration flow paths for the PCIV Position Function. As such, a Note has been provided that allows separate Condition entry for each inoperable PAM Function, including separate Condition entry for each penetration flow path for the PCIV Position Function.

A.1

When one or more Functions have one required channel that is inoperable, the required inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channels (or, in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the

BASES

APPLICABILITY

The Alternate Shutdown System LCO is applicable in MODES 1 and 2. This is required so that the plant can be placed and maintained in a safe shutdown condition for an extended period of time from a location other than the control room.

This LCO is not applicable in MODES 3, 4, and 5. In these MODES, the plant is already subcritical and in a condition of reduced Reactor Coolant System energy. Under these conditions, considerable time is available to restore necessary instrument control Functions if control room instruments or control becomes unavailable. Consequently, the LCO does not require OPERABILITY in MODES 3, 4, and 5.

ACTIONS

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

A.1

Condition A addresses the situation where one or more required Functions of the Alternate Shutdown System is inoperable. This includes any Function listed in Table B 3.3.3.2-1.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

In general, the individual Functions are required to be OPERABLE in MODES 1, 2, and 3 consistent with the Applicability for LCO 3.6.1.1, "Primary Containment." Functions that have different Applicabilities are discussed below in the individual Functions discussion.

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

Main Steam Line Isolation

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

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. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level — Low Low Low (Level 1) Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level — Low Low Low (Level 1) Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 4). The isolation of the MSLs on Level 1 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four level transmitters 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. Four channels of Reactor Vessel Water Level — Low Low Low (Level 1) 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 Low (Level 1) Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 50.67 limits.

This Function isolates the MSIVs and MSL drains.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Primary Containment Isolation

2.a. Reactor Vessel Water Level - Low (Level 3)

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 3 supports actions to ensure that offsite dose limits of 10 CFR 50.67 are not exceeded. The Reactor Vessel Water Level — Low (Level 3) Function associated with isolation is implicitly assumed in the USAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level — Low (Level 3) signals are initiated from four vessel level instrument 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. Four channels of Reactor Vessel Water Level — Low (Level 3) 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 (Level 3) Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group 2, 3, and 6 valves listed in Reference 1.

2.b. Drywell Pressure - High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 50.67 are not exceeded. The Drywell Pressure — High Function, associated with isolation of the primary containment, is implicitly assumed in the USAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

from the reactor vessel occurs as part of the isolation of Primary Containment to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level - Low Low Low (Level 1) Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. This Function, associated with Primary Containment isolation, is assumed in the analysis of the recirculation line break (Ref. 4). The isolation of the valves of the recirculation sample line on Level 1 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four level transmitters 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. Four channels of Reactor Vessel Water Level - Low Low Low (Level 1) 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 Low (Level 1) Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the recirculation sample valves will isolate on a potential LOCA to prevent offsite doses from exceeding 10 CFR 50.67 limits.

This Function isolates the recirculation sample valves. It may be bypassed using a key-locked switch during accident conditions to obtain a sample for Post Accident Sampling System (PASS).

<u>High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems Isolation</u>

<u>3.a., 3.b., 4.a., 4.b. HPCI and RCIC Steam Line Flow - High and Time Delay Relays</u>

Steam Line Flow — High Functions are provided to detect a break of the RCIC or HPCI steam lines and initiate closure of the steam line isolation valves of the appropriate system. If the steam is allowed to continue flowing out of the break, the reactor will depressurize and the core can uncover. Therefore, the isolations are initiated on high flow to prevent or minimize core damage. The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. Specific credit for these Functions is not assumed in any USAR accident analyses since the bounding analysis is performed for large breaks such as recirculation and

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILTY (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.

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

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 or 10 CFR 50.67 (Fuel Handling Accident and LOCA only).

BASES

LCO (continued)

the leakage detection systems inoperable, monitoring for LEAKAGE in the RCPB is degraded.

APPLICABILITY

In MODES 1, 2, and 3, leakage detection systems are required to be OPERABLE to support LCO 3.4.4. This Applicability is consistent with that for LCO 3.4.4.

ACTIONS

A.1

With the drywell floor drain sump flow monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, the drywell atmospheric activity monitor will provide indication of changes in leakage.

With the drywell floor drain sump flow monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every 12 hours (SR 3.4.4.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still available.

B.1 and B.2

With all three of the gaseous and particulate drywell atmospheric monitoring channels inoperable, grab samples of the drywell atmosphere must be taken and analyzed to provide periodic leakage information. Provided a sample is obtained and analyzed once every 12 hours, the plant may be operated for up to 30 days to allow restoration of at least one of the required monitors.

The 12 hour interval provides periodic information that is adequate to detect LEAKAGE. The 30 day Completion Time for

BASES

ACTIONS (continued)

restoration recognizes that at least one other form of leakage detection is available.

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B cannot be met, 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 MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the actions in an orderly manner and without challenging plant systems.

D.1

With all required monitors inoperable, no required automatic means of monitoring LEAKAGE are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR is for the performance of a CHANNEL CHECK of the required drywell atmospheric monitoring system. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the reactor coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the reactor coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure that in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR 100 (Ref. 1) for the Control Rod Drop and Main Steam Link Break accidents, and 10 CFR 50.67, "Accident Source Term," (Ref. 6) for the Fuel Handling and Loss-of-Coolant accidents.

This LCO contains iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to a small fraction of the 10 CFR 100 limit for the Control Rod Drop and Main Steam Line Break accidents, and within the 10 CFR 50.67, "Accident Source Term," (Ref. 6) limit for the Fuel Handling and Loss-of-Coolant accidents.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the USAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

This MSLB release forms the basis for determining offsite and control room doses (Refs. 2 and 3). The limits on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses at the site boundary,

BASES

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 0.2 μCi/gm within 48 hours, or if at any time it is > 4.0 μCi/gm, it must be determined at least once every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR 100 during a postulated MSLB accident.

Alternatively, the plant can be placed in MODE 3 within 12 hours and in MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

BASES

ACTIONS (continued)

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for placing the unit in MODES 3 and 4 are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. 10 CFR 100.11, 1973.
2. USAR, Section XIV-8.1.
3. USAR, Section XIV-6.5.
4. 10 CFR 50, Appendix A, GDC 19.
5. 10 CFR 50.36(c)(2)(ii).
6. 10 CFR 50.67.

BASES

APPLICABILITY (continued)

typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS — Operating") do not allow placing the RHR shutdown cooling subsystem into operation.

The requirements for decay heat removal in MODES 4 and 5 are discussed in LCO 3.4.8, "Residual Heat Removal (RHR) Shutdown Cooling System — Cold Shutdown"; LCO 3.9.7, "Residual Heat Removal (RHR) — High Water Level"; and LCO 3.9.8, "Residual Heat Removal (RHR) — Low Water Level."

ACTIONS

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

A.1, A.2, and A.3

With one required RHR shutdown cooling subsystem inoperable for decay heat removal, except as permitted by LCO Note 2, the inoperable subsystem must be restored to OPERABLE status

BASES

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."

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable HPCI system. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable HPCI system and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.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.

BASES

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable RCIC system. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable RCIC system and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1 and A.2

If the RCIC System is inoperable during MODE 1, or MODE 2 or 3 with reactor steam dome pressure > 150 psig, and the HPCI System is verified to be OPERABLE, the RCIC System must be restored to OPERABLE status within 14 days. In this Condition, loss of the RCIC System will not affect the overall plant capability to provide makeup inventory at high reactor pressure since the HPCI System is the only high pressure system assumed to function during a loss of coolant accident (LOCA). OPERABILITY of HPCI is therefore verified within 1 hour when the RCIC System is inoperable. This may be performed as an administrative check, by examining logs or other information, to determine if HPCI is out of service for maintenance or other reasons. It does not mean it is necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the HPCI System. If the OPERABILITY of the HPCI System cannot be verified, however, Condition B must be immediately entered. For transients and certain abnormal events with no LOCA, RCIC (as opposed to HPCI) is the preferred source of makeup coolant because of its relatively small capacity, which allows easier control of the RPV water level. Therefore, a limited time is allowed to restore the inoperable RCIC to OPERABLE status.

The 14 day Completion Time is consistent with the recommendations in a reliability study (Ref. 4) 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 outage times (AOTs). Because of similar functions of HPCI and RCIC, the AOTs (i.e., Completion Times) determined for HPCI are also applied to RCIC.

BASES

ACTIONS (continued)

B.1 and B.2

If the RCIC System cannot be restored to OPERABLE status within the associated Completion Time, or if the HPCI System is simultaneously inoperable, the plant must be brought to a condition 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 reactor steam dome pressure reduced to ≤ 150 psig within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The flow path piping has the potential to develop voids and pockets of entrained air. Maintaining the pump discharge line of the RCIC System full of water ensures that the system will perform properly, injecting its full capacity into the Reactor Coolant System upon demand. This will also prevent a water hammer following an initiation signal. One acceptable method of ensuring the line is full is to vent at the high points. The 31 day Frequency is based on the gradual nature of void buildup in the RCIC piping, the procedural controls governing system operation, and operating experience.

SR 3.5.3.2

Verifying the correct alignment for manual, power operated, and automatic valves in the RCIC flow path provides assurance that the proper flow path will exist for RCIC operation. This SR applies only to valves affecting the direct flow path. This SR excludes 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

BASES

SURVEILLANCE REQUIREMENTS (continued)

valves that cannot be inadvertently misaligned, such as check valves. For the RCIC System, this SR also includes the steam flow path for the turbine and the flow controller position.

The 31 day Frequency of this SR was derived from the Inservice Testing Program requirements for performing valve testing at least once every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control and because improper valve position would affect only the RCIC System. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.3.3 and SR 3.5.3.4

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow against a system head corresponding to reactor pressure is tested both at the higher and lower operating ranges of the system. The required system head should overcome the RPV pressure and associated discharge line losses. Adequate reactor steam pressure must be available to perform these tests. 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. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these SRs. Adequate reactor steam pressure to perform SR 3.5.3.3 is 920 psig and 145 psig to perform SR 3.5.3.4. Adequate steam flow is represented by turbine bypass valves at least 30% open, or total steam flow $\geq 10^6$ lb/hr. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure Surveillance has been satisfactorily completed and there is no indication or reason to believe that RCIC is inoperable. Therefore, these SRs are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are

BASES

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in References 1 and 2. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

The maximum allowable leakage rate for the primary containment (L_a) is 0.635% by weight of the containment air per 24 hours at the accepted value for the design basis LOCA peak containment pressure (P_a) of 58 psig (Ref. 5).

Primary containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

Primary containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met. In addition, the leakage from the drywell to the suppression chamber must be limited to ensure the pressure suppression function is accomplished and the suppression chamber pressure does not exceed design limits. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is

BASES

SURVEILLANCE REQUIREMENTS (continued)

does not change by more than the calculated amount per minute over a 10 minute period. The leakage test is performed every 18 months. The 18 month Frequency was developed considering it is prudent that this Surveillance be performed during a unit outage and also in view of the fact that component failures that might have affected this test are identified by other primary containment SRs. Two consecutive test failures, however, would indicate unexpected primary containment degradation; in this event, as the Note indicates, increasing the Frequency to once every 9 months is required until the situation is remedied as evidenced by passing two consecutive tests.

REFERENCES

1. USAR, Section V-2.4.
2. USAR, Section XIV-6.3.
3. 10 CFR 50, Appendix J, Option B.
4. 10 CFR 50.36(c)(2)(ii).
5. Safety Evaluation Report by U.S. Atomic Energy Commission dated February 14, 1973 (Section 6.2.1).

BASES

BACKGROUND (continued)

containment leakage rate to within limits in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the safety analysis.

APPLICABLE SAFETY ANALYSES

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 0.635% by weight of the containment air per 24 hours at the accepted value for the calculated peak containment pressure (P_a) of 58 psig (Ref. 2). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment.

The primary containment air lock satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii) (Ref. 3).

LCO

As part of the primary containment pressure boundary, the air lock's safety function is related to control of containment leakage following a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

The primary containment air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door to be opened at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be

BASES

SURVEILLANCE REQUIREMENTS (continued)

successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR, requiring the results to be evaluated against the acceptance criteria which are applicable to SR 3.6.1.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C primary containment leakage rate.

SR 3.6.1.2.2

The air lock interlock mechanism is designed to prevent simultaneous opening of both doors in the air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident primary containment pressure, closure of either door will support primary containment OPERABILITY. Thus, the interlock feature supports primary containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when primary containment is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the potential for loss of primary containment OPERABILITY if the Surveillance were performed with the reactor at power. The 24 month Frequency is based on engineering judgement and is considered adequate given that the interlock is not challenged during use of the airlock.

REFERENCES

1. USAR, Section V-2.3.4.5.

2. Safety Evaluation Report by U.S. Atomic Energy Commission dated February 14, 1973 (Section 6.2.1).

3. 10 CFR 50.36(c)(2)(ii).

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.9

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. The Frequency of 18 months on a STAGGERED TEST BASIS is considered adequate given the administrative controls on replacement charges and the frequent checks of circuit continuity (SR 3.6.1.3.4).

SR 3.6.1.3.10

The analyses in References 8 and 9 are based on leakage that is less than the specified leakage rate. A leakage rate of 150 scfh per Main Steam line at $\geq P_a$ (58 psig) was assumed in the LOCA analyses. The equivalent leakage rate at $\geq P_t$ (29 psig) is 106 scfh. An "MSIV line" is each one of the four Main Steam lines with an inboard and an outboard Main Steam Isolation Valve (MSIV). The leakage rate to be measured is the Main Steam line "minimum path" leakage (the lesser actual pathway leakage of the two MSIVs in the Main Steam Line). The leakage limit is based on the analyses of References 11 and 12. The Frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.11

Verifying each inboard 24 inch primary containment purge and vent valve (PC-230 MV, PC-231 MV, PC-232 MV, and PC-233 MV) is blocked to restrict the maximum opening angle to 60° is required to ensure that the valves can close under DBA conditions within the times assumed in the analysis of References 7 and 8. If a LOCA occurs, the purge and vent valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times, pressurization concerns are not present, thus the purge valves can be fully open. The 18 month Frequency is appropriate because the blocking devices may be removed during a refueling outage.

SR 3.6.1.3.12

The Main Steam Pathway is the analyzed leakage path from the four Main Steam lines and the inboard Main Steam drain line to and including the condenser. The leakage limit imposed on the Main Steam Pathway with this surveillance requirement applies to the total (aggregate) leakage for the Main Steam Pathway. The Main Steam Pathway leakage includes the total leakage of all four Main Steam line penetrations plus

BASES

SURVEILLANCE REQUIREMENTS (continued)

the inboard Main Steam drain line penetration (penetration X-8, with Containment isolation valves MS-MOV-MO74 and MS-MOV-MO77). The aggregate leakage of the Main Steam Pathway must be ≤ 212 scfh when tested at ≥ 29 psig or the equivalent leakage rate of ≤ 300 scfh when tested at ≥ P_a (58 psig). The leakage rate to be measured is the Main Steam Pathway "minimum path" leakage (the lesser actual pathway leakage of the two Containment isolation valves in each penetration; i.e., the MSIVs in penetrations X-7A through X-7D, and MS-MOV-MO74 and MS-MOV-MO77 in penetration X-8). The leakage limit is based on the analyses of References 11 and 12. The Frequency is specified in the Primary Containment Leakage Rate Testing Program.

REFERENCES

1. USAR, Chapter XIV.
2. Amendment 25 to the FSAR.
3. NEDC 96-006, "Estimate of Steam Tunnel's HELB," dated March 30, 1996.
4. USAR Section IV-4.9.
5. 10 CFR 50.36(c)(2)(ii).
6. USAR, Table V-2-2.
7. USAR, Burns and Roe Drawing 4259, Sheets 1 and 1A, and Burns and Roe Drawing 4260, Sheets 2A and 2B (Incorporated by Reference).
8. USAR, Section V-2.0.
9. USAR, Section XIV-6.3.
10. 10 CFR 50, Appendix J, Option B.
11. NEDC 07-082, "Radiological Dose Analysis for a Loss of Coolant Accident (LOCA) at Cooper Nuclear Station."
12. NEDC 07-087, "AST Reduced Pressure MS Pathway Leakage Limits."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Pressure

BASES

BACKGROUND

The drywell pressure is limited during normal operations to preserve the initial conditions assumed in the accident analysis for a Design Basis Accident (DBA) or loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for the entire spectrum of break sizes for postulated LOCAs (Ref. 1). Among the inputs to the DBA is the initial primary containment internal pressure (Ref. 1). Analyses assume an initial drywell pressure of 0.75 psig. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell internal pressure does not exceed the maximum allowable of 62 psig.

The maximum calculated drywell pressure occurs during the reactor blowdown phase of the DBA, which assumes an instantaneous recirculation line break. The accepted value for the calculated peak drywell pressure for this limiting event is 58 psig (Ref. 3).

Drywell pressure satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii) (Ref. 2).

LCO

In the event of a DBA, with an initial drywell pressure $\leq$ 0.75 psig, the resultant peak drywell accident pressure will be maintained below the maximum allowable drywell pressure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining drywell pressure within limits is not required in MODE 4 or 5.

BASES

ACTIONS

A.1

With drywell pressure not within the limit of the LCO, drywell pressure must be restored within 1 hour. The Required Action is necessary to return operation to within the bounds of the primary containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, "Primary Containment," which requires that primary containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If drywell pressure cannot be restored to within limit within the required 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.4.1

Verifying that drywell pressure is within limit ensures that unit operation remains within the limit assumed in the primary containment analysis. The 12 hour Frequency of this SR was developed, based on operating experience related to trending of drywell pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal drywell pressure condition.

REFERENCES

1. USAR, Section V-2.

2. 10 CFR 50.36(c)(2)(ii).

3. Safety Evaluation Report by the U.S. Atomic Energy Commission dated February 14, 1973 (Section 6.2.1).

B 3.7 PLANT SYSTEMS

B 3.7.2 Service Water SW System and Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The SW System is designed to provide cooling water for the removal of heat from equipment, such as the diesel generators (DGs) and Reactor Equipment Cooling (REC) System heat exchangers, and to provide a supply of water for the Residual Heat Removal Service Water Heat Exchangers through the Residual Heat Removal Service Water Booster (RHRSWB) System pumps, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The SW System also provides cooling to unit components, as required, during normal operation. The SW System also provides cooling water to turbine building non-essential loads. If SW header pressure falls below 20 psig, the system automatically isolates the non-essential header by closing the discharge cross-tie valves. The SW system can be manually aligned as a backup to the REC System through remotely controlled motor operated valves. This configuration would be used in the event that the REC System becomes incapable of performing its essential cooling function and in this configuration the SW System provides cooling water to the room coolers for the Emergency Core Cooling System (Core Spray, RHR, HPCI) pump rooms and the RHR pump seal water coolers.

The SW System consists of the Ultimate Heat Sink (UHS) and two independent and redundant subsystems. Each of the two SW subsystems is made up of a header, two 8000 gpm pumps, a suction source, valves, piping and associated instrumentation. Either of the two subsystems is capable of providing the required cooling capacity to support the required systems with one pump operating. The two subsystems are separated from each other so failure of one subsystem will not affect the OPERABILITY of the other system.

Cooling water is pumped from the Missouri River by the SW pumps to the essential components through the two main headers. After removing heat from the components, the water is collected into two discharge headers and routed to the discharge canal where the water is returned to the river. Service Water discharge from the turbine equipment cooling (TEC) heat exchangers is routed to the 1A circulating water (CW) discharge tunnel.

BASES

APPLICABLE SAFETY ANALYSES

Sufficient water level is available for all SW System post LOCA cooling requirements when the river level is at least 865 ft mean sea level with no additional makeup water source available (Ref. 1). A river level of 865 ft mean sea level equates to a level of at least 863.2 ft mean sea level in the SW pump bay under postulated worst case conditions. This level exceeds the 862.8 ft mean sea level submergence requirements for necessary long term SW cooling. The ability of the SW System to support long term cooling of the reactor containment is assumed in evaluations of the equipment required for safe reactor shutdown presented in the USAR, Chapters V and XIV (Refs. 2 and 3, respectively). These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

The ability of the SW System to provide adequate cooling to the identified safety equipment is an implicit assumption for the safety analyses evaluated in References 2 and 3. The ability to provide onsite emergency AC power is dependent on the ability of the SW System to cool the DGs. The long term cooling capability of the RHR, core spray, and RHRSWB pumps is also dependent on the cooling provided by the SW System. SW backup to REC is capable of performing the essential heat removal function as evaluated in Reference 5.

The SW System, together with the UHS, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

The SW subsystems are independent of each other to the degree that each has separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, one subsystem of SW is required to provide the minimum heat removal capability assumed in the safety analysis for the system to which it supplies cooling water. To ensure this requirement is met, two subsystems of SW must be OPERABLE. At least one subsystem will operate, if the worst single active failure occurs coincident with the loss of offsite power.

A subsystem is considered OPERABLE when it has an OPERABLE UHS, two OPERABLE pumps, and an OPERABLE flow path capable of taking suction from the intake structure and transferring the water to the appropriate equipment.

The OPERABILITY of the UHS is based on having a minimum river water level of 865 ft mean sea level and a maximum water temperature of 95°F.

The isolation of the SW System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the SW System.

BASES

APPLICABILITY

In MODES 1, 2, and 3, the SW System and UHS are required to be OPERABLE to support OPERABILITY of the equipment serviced by the SW System. Therefore, the SW System and UHS are required to be OPERABLE in these MODES.

Under other plant conditions, the OPERABILITY requirements of the SW System and UHS are determined by the systems they support and therefore, the requirements are not the same for all facets of operation. Thus, the LCOs of the RHR Shutdown Cooling System (LCO 3.4.8, "RHR Shutdown Cooling System—Cold Shutdown," LCO 3.5.2, "ECCS—Shutdown," LCO 3.8.2, "AC Sources—Shutdown," LCO 3.9.7, "RHR—High Water Level," and LCO 3.9.8, "RHR—Low Water Level"), which require portions of the SW System to be OPERABLE, will govern SW System operation in MODES 4 and 5.

ACTIONS

A.1

With one SW subsystem inoperable, the SW subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE SW subsystem is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SW subsystem could result in loss of SW function.

The 30 day Completion Time is based on the redundant SW System capabilities afforded by the OPERABLE subsystem and the low probability of an accident occurring during this time period.

Required Action A.1 is modified by two Notes indicating that the applicable Conditions of LCO 3.8.1, "AC Sources — Operating," LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System — Hot Shutdown," be entered and Required Actions taken if the inoperable SW subsystem results in an inoperable DG or RHR shutdown cooling subsystem, respectively. This is in accordance with LCO 3.0.6 and ensures the proper actions are taken for these components.

B.1 and B.2

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies the river water level to be sufficient for the proper operation of the SW pumps (net positive suction head and pump vortexing are considered in determining this limit). The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.2.2

Verification of the UHS temperature ensures that the heat removal capability of the SW System is within the assumptions of the DBA analysis. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.2.3

Verifying the correct alignment for each manual, power operated, and automatic valve in each SW subsystem flow path provides assurance that the proper flow paths will exist for SW operation. This SR applies only to valves affecting the direct flow path. This SR excludes 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 is also allowed to be in the nonaccident position, and yet considered in the correct position, provided it can be automatically realigned to its accident position within the required time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

This SR is modified by a Note indicating that isolation of the SW System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the SW System. As such, when all SW pumps, valves, and piping are OPERABLE, but a branch connection off the main header is isolated, the SW System is still OPERABLE.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.4

This SR verifies that the automatic isolation valves of the SW System will automatically switch to the safety or emergency position to provide cooling water exclusively to the safety related equipment during an accident event. This is demonstrated by the use of an actual or simulated initiation signal. The initiation signal is caused by low SW header pressure (approximately 20 psig). This SR also verifies the automatic start capability of one of the two SW pumps in each subsystem.

Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

REFERENCES

1. NEDC 94-255, "Hydraulic Evaluation of Opening in Intake Structure Guide Wall," June 14, 1995.
2. USAR, Chapter V.
3. USAR, Chapter XIV.
4. 10 CFR 50.36(c)(2)(ii).
5. NEDC 00-095E, "CNS Reactor Building Post-LOCA Heating Analysis," October 22, 2007.

B 3.7 PLANT SYSTEMS

B 3.7.3 Reactor Equipment Cooling (REC) System

BASES

BACKGROUND

The REC System is designed to provide cooling water for the removal of heat from equipment, such as the room coolers for the core spray pump rooms, RHR pump rooms, and HPCI pump room, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient and the RHR pump seal water coolers (essential loads). The REC System also provides cooling to unit components, as required, during normal operation (non-essential loads). In the event of a loss of REC System pressure, automatic valving is provided to shut off all supply to nonessential loads, thus assuring supply to the essential loads.

The REC System consists of two subsystems, each consisting of two 1350 gpm pumps, a heat exchanger, valves, piping and associated instrumentation. A 550 gallon capacity surge tank, located at the highest point of the system, accommodates system volume changes, maintains static pressure in the loops to ensure adequate net positive suction head (NPSH), detects gross leaks in the REC System by providing a point to monitor inventory, and provides a means for adding water. Either of the two subsystems with one REC pump operating or the Service Water supply, is capable of providing the required cooling capacity to support the required essential systems. The two subsystems have sufficient redundancy and independence from each other such that no active component failure in one subsystem will affect the OPERABILITY of the other. Additionally, each subsystem is provided with Service Water supply and return valves to provide required component cooling in the event of REC leakage in excess of limits or a passive failure, such as a Class 1E pipe break.

The Service Water (SW) System provides two subsystems for backup to the REC critical loops. The Service Water supply is a fully qualified essential supply to the REC critical loops. This configuration would be used in the event that the REC system becomes incapable of performing its essential cooling function. Because of the cross-tie capability in the critical REC loops either SW backup subsystem can supply the required cooling to the REC System.

Cooling water is pumped by the REC pumps, delivered to the REC heat exchangers, which are cooled by the Service Water System, and then to the components through the two main headers. After removing heat from the components, the water is then recirculated back to the REC pump suction.

BASES

APPLICABLE SAFETY ANALYSIS

Either REC loop has sufficient capacity with one pump operating to transfer the essential services design cooling heat load during postulated transient or accident conditions (Ref. 1). However, to provide additional margin, two REC pumps per loop are required to be OPERABLE to satisfy the requirements of the LCO.

Through the intertie with the REC System, the SW System provides essential cooling equivalent to the critical loops of the REC System. The ability of the REC System, or the associated service water supply, to provide adequate cooling to the identified safety equipment is an implicit assumption for the safety analyses evaluated in Reference 1.

The REC System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

LCO

The REC subsystems are independent of each other to the degree that each has separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, one subsystem of REC is required to provide the minimum heat removal capability assumed in the safety analysis for the system to which it supplies cooling water. To ensure this requirement is met, two subsystems of REC must be OPERABLE. At least one subsystem will operate, if the worst single active failure occurs coincident with the loss of offsite power.

A subsystem is considered OPERABLE when it has two OPERABLE pumps, one OPERABLE heat exchanger, and an OPERABLE flow path capable of transferring the water to the appropriate equipment.

The OPERABILITY of the REC System is based on verifying a maximum supply water temperature of 100°F.

An REC subsystem is considered inoperable if both of the following conditions exits: (1) leakage in excess of allowable limits, and (2) the subsystem of SW backup for the respective REC subsystem is inoperable. The limits are based on having a 30-day supply of inventory in the REC surge tank without crediting makeup. Leakage in excess of limits by itself does not result in either the REC subsystems or the REC system being inoperable. If it is determined that leakage exceeds these limits, then REC is considered degraded and SW backup is required to be OPERABLE to maintain the REC subsystem(s) OPERABLE. An OPERABLE SW backup subsystem requires an OPERABLE flow path from the SW System to the REC critical loops, an OPERABLE SW pump in the respective SW Subsystem, and the ability to align the SW backup valves in the REC System.

BASES

LCO (continued)

The isolation of the REC System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the REC System.

APPLICABILITY

In MODES 1, 2, and 3, the REC System is required to be OPERABLE to support OPERABILITY of the equipment serviced by the REC System. Therefore, the REC System is required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the REC System are determined by the systems it supports.

ACTIONS

A.1

Both REC subsystems are considered degraded if REC leakage exceeds limits. If REC leakage exceeds limits, in combination with one of the SW backup subsystems being inoperable, the REC System is considered OPERABLE but degraded and the remaining SW backup subsystem must be verified to be OPERABLE within 1 hour. The limit on REC leakage is based on maintaining 30 days of inventory in the surge tank without makeup to the surge tank. Administrative means may be used to verify the remaining SW backup subsystem is OPERABLE.

A.2.1 and A.2.2

These actions require returning the inoperable SW backup system to OPERABLE status or restoring REC leakage to within limits within 14 days. The overall reliability is reduced in this condition since a single failure in the OPERABLE SW backup subsystem could result in a loss of the REC heat removal function.

The 14-day Completion Time is based on the plant risk being lower during this period of continued plant operation than the risk of shutting down and on the REC System remaining OPERABLE but degraded with REC leakage exceeding limits and one OPERABLE SW backup system during this period.

BASES

ACTIONS (continued)

B.1

With one REC subsystem inoperable for reasons other than Condition A, the REC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE REC subsystem is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE REC subsystem could result in loss of REC function.

The 30 day Completion Time is based on the redundant REC System capabilities afforded by the OPERABLE subsystem and the low probability of an accident occurring during this time period.

C.1 and C.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies the water level in the REC surge tank to be sufficient for the proper operation of the REC System (system volume changes, static pressure in the loops, and potential leakage in the system are considered in determining this limit). If REC leakage exceeds limits, the REC subsystems are considered OPERABLE but degraded. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

This SR is modified by two Notes. Note 1 states that SR 3.0.1 is not applicable when both SW backup subsystems are OPERABLE. Note 2 states that REC leakage beyond limits by itself is only a degraded condition and does not render the REC System inoperable. These notes reflect that the REC System remains OPERABLE based on the ability to align the SW System to the REC System and supply the required cooling water to the critical loops of the REC System.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.3.2

Verification of the REC System temperature ensures that the heat removal capability of the REC System is within the assumptions of the DBA analysis. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.3.3

Verifying the correct alignment for each manual, power operated, and automatic valve in each REC subsystem flow path provides assurance that the proper flow paths will exist for REC operation. 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 is also allowed to be in the nonaccident position, and yet considered in the correct position, provided it can be automatically realigned to its accident position within the required time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

This SR is modified by a Note indicating that isolation of the REC System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the REC System. As such, when all REC pumps, valves, and piping are OPERABLE, but a branch connection off the main header is isolated, the REC System is still OPERABLE.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.3.4

This SR verifies that the automatic isolation valves of the REC System will automatically switch to the safety or emergency position to provide cooling water exclusively to the safety related equipment during an accident event. This is demonstrated by the use of an actual or simulated initiation signal. The initiation signal is caused by low REC heat exchanger outlet pressure (which has an analytically determined limit of 55 psig decreasing). Also, a Group VI isolation signal will open

BASES

SURVEILLANCE REQUIREMENTS (continued)

the REC heat exchanger service water outlet valves and the REC critical loop supply valves to provide cooling water to essential components.

Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section X-6.
2. 10 CFR 50.36(c)(2)(ii).
3. DC 93-057
4. NEDC 92-050X and NEDC 97-087

B 3.7 PLANT SYSTEMS

B 3.7.4 Control Room Emergency Filter System

BASES

BACKGROUND

The CREF System provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke.

The safety related function of the CREF System includes a single high efficiency air filtration system for emergency treatment of outside supply air and a CRE boundary that limits the inleakage of unfiltered air. The system consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a supply fan, an emergency booster fan, an exhaust booster fan, and the associated ductwork, valves or dampers, doors, barriers, and instrumentation. Prefilters and HEPA filters remove particulate matter, which may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations, and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the leakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) (loss-of-coolant and fuel handling accidents only) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The CREF System is a standby system. Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to CRE occupants), the CREF System automatically switches to the emergency bypass mode of operation to minimize infiltration of contaminated air into the CRE. A system of dampers isolates the normal outside air intake path, and the outside air is routed through the filter system. Outside air is taken in at the normal ventilation intake.

BASES

BACKGROUND (continued)

The CREF System is designed to maintain a habitable environment in the CRE for a 30-day continuous occupancy after a DBA without exceeding 5 rem whole body dose or its equivalent to any part of the body following a loss-of-coolant accident (LOCA) (Ref. 8) or 5 rem total effective dose equivalent (TEDE) following a fuel handling accident (FHA). The CREF System will pressurize the CRE relative to external areas adjacent to the CRE boundary to minimize infiltration of air from all surrounding areas adjacent to the CRE boundary. CREF System operation in maintaining CRE habitability is discussed in the USAR, Chapters X and XIV, (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The ability of the CREF System to maintain the habitability of the CRE is an explicit assumption for the safety analyses presented in the USAR, Chapters X and XIV (Refs. 1 and 2, respectively). The CREF System is assumed to operate following a loss of coolant accident and a fuel handling accident involving handling lately irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 7 days).

The CRE boundary provides protection from smoke and hazardous chemicals to the CRE occupants by manual isolation after detecting the presence of smoke and chemicals, thereby minimizing the quantity of inleakage. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release. The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels (Ref. 5).

The CREF System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

The CREF System is required to be OPERABLE, since total system failure, such as from an inoperable CRE boundary, could result in exceeding a dose of 5 rem whole body dose or its equivalent to any part of the body following a LOCA or 5 rem TEDE following a FHA to the CRE occupants.

The CREF System is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. The system is considered OPERABLE when its associated:

BASES

LCO (continued)

a. Fans are OPERABLE (one supply fan, the emergency booster fan and the exhaust booster fan);

b. HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In order for the CREF System to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke. The CRE boundary protects CRE occupants from sources of hazardous chemicals and smoke external to the CRE by minimizing inleakage when isolated. Because the CRE boundary provides no protection from internal sources, the CRE occupants are protected from internal sources by donning SCBAs.

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. Application of the Note allows the LCO to be considered as met because the boundary can be restored to its design condition by quickly closing the door, hatch, floor plug, or access panel. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For openings such as blocked-open doors, hatches, plugs, or access panels, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated. For any other openings, such as holes and removed doors, the same controls should be implemented; however, the LCO is considered as not met, and the Conditions and Required Actions are to be entered.

BASES

APPLICABILITY

In MODES 1, 2, and 3, the CREF System must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA, since the DBA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the CREF System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During operations with a potential for draining the reactor vessel (OPDRVs); and

b. During movement of lately irradiated fuel assemblies in the secondary containment. Due to radioactive decay, the CREF System is only required to be OPERABLE during fuel handling involving handling lately irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 7 days).

ACTIONS

A.1

When inoperable for reasons other than an inoperable CRE boundary, the inoperable CREF System must be restored to OPERABLE status within 7 days. With the unit in this condition, there is no other system to perform the CRE occupant protection function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period.

B.1, B.2, and B3

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem whole body or its equivalent to any part of the body following a LOCA or 5 rem TEDE following a FHA), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Protection of CRE occupants from hazardous chemicals or smoke is inadequate if they are unable to remain in the CRE and perform their duties. The CRE boundary must be restored to OPERABLE status within 90 days.

BASES

ACTIONS (continued)

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke, i.e., the CRE occupants are able to remain in the CRE and perform their duties. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan, and possibly repair, and test most problems with the CRE boundary.

C.1 and C.2

In MODE 1, 2, or 3, if the inoperable CREF System or the CRE boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1, and D.2

The Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving lately irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of lately irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

ACTIONS (continued)

During movement of lately irradiated fuel assemblies in the secondary containment or during OPDRVs, if the inoperable CREF System cannot be restored to OPERABLE status within the required Completion Time, or with the CREF System inoperable due to an inoperable CRE boundary, activities that present a potential for releasing radioactivity that might require isolation of the CRE must be immediately suspended. This places the unit in a condition that minimizes the accident risk.

If applicable, movement of lately irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the CREF System in a standby mode starts on demand and continues to operate. The system should be checked periodically to ensure that it starts and functions properly. As the environmental and normal operating conditions of this system are not severe, testing the system once every month provides an adequate check on this system. Since the CREF System does not contain heaters, the system need only be operated for $\geq$ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the known reliability of the equipment.

SR 3.7.4.2

This SR verifies that the required CREF 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.4.3

This SR verifies that on an actual or simulated initiation signal, the CREF System starts and operates. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.7.1, "Control Room Emergency Filter (CREF) System Instrumentation," overlaps this SR to provide complete testing of the safety function. The Frequency of 18 months is based on industry operating experience and is consistent with the typical refueling cycle.

SR 3.7.4.4

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of this testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem whole body or its equivalent to any part of the body following a LOCA or 5 rem TEDE following a FHA and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 4) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 6). These compensatory measures may also be used as mitigating actions as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 7). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

BASES

REFERENCES

1. USAR, Chapter X.

2. USAR, Chapter XIV.

3. 10 CFR 50.36(c)(2)(ii).

4. Regulatory Guide 1.196.

5. USAR, Section X-10.4.6.5.

6. NEI 99-03, "Control Room Habitability Assessment Guidance," June 2001.

7. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability." (ADAMS Accession No. ML040160868).

8. Standard Review Plan (NUREG-0800, Rev. 2), Table 6.4-1, from Section 6.4.

B 3.7 PLANT SYSTEMS

B 3.7.5 Air Ejector Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the main condenser. Air and noncondensible gases are collected in the main condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Air Ejector Offgas System, where they undergo a hold-up period of 30 minutes, and are then exhausted to the Augmented Offgas Treatment System. The offgas from the main condenser normally includes radioactive gases.

The Augmented Offgas Treatment System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled and dried to reduce the moisture content, passed through charcoal beds for delay and decay of noble gas activity, and exhausted through the Elevated Release Point (ERP) with proper dilution. The radioactivity of the gaseous mixture exiting through the ERP is monitored by the ERP Exhaust Monitor Air Sampling System.

APPLICABLE SAFETY ANALYSES

A gross gamma activity rate limit of 0.39 Ci/sec noble gas release rate at the air ejectors (after 30 minute decay) is used as the source term for the accident analysis of the Augmented Offgas Treatment System. Using this source term, the maximum offsite total body dose would be < 0.5 rem (Ref. 1). However, a more restrictive limit of 1.0 Ci/sec (prior to 30 minute decay) has been established to provide an improved capability to detect fuel pin cladding failures to allow prevention of serious degradation of fuel pin cladding integrity which might result from plant operation with a misoriented or misloaded fuel assembly. Therefore, the gross gamma activity rate ensures that serious degradation of fuel pin cladding integrity can be detected, while maintaining the calculated offsite doses well within the limits of 10 CFR 100 (Ref. 2).

The air ejector offgas limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

To ensure compliance with the accident analysis for the Augmented Offgas Treatment System and to provide improved capability to detect fuel pin cladding failures, gross gamma activity rate of noble gases shall be ≤ 1.0 Ci/sec (prior to 30 minute decay).

BASES (continued)

APPLICABILITY

The LCO is applicable when steam is being exhausted to the main condenser and the resulting noncondensibles are being processed via the Air Ejector Offgas System. This occurs during MODE 1, and during MODES 2 and 3 with any main steam line not isolated and the SJAE in operation. In MODES 4 and 5, main steam is not being exhausted to the main condenser and the requirements are not applicable.

ACTIONS

A.1

If the offgas radioactivity rate limit is exceeded, 72 hours is allowed to restore the gross gamma activity rate to within the limit. The 72 hour Completion Time is reasonable, based on engineering judgment, the time required to complete the Required Action, the large margins associated with permissible dose and exposure limits, and the low probability of an Air Ejector Offgas System rupture.

B.1, B.2, B.3.1, and B.3.2

If the gross gamma activity rate is not restored to within the limits in the associated Completion Time, all main steam lines or the SJAE must be isolated. This isolates the Air Ejector Offgas System from significant sources of radioactive steam. The main steam lines are considered isolated if at least one main steam isolation valve in each main steam line is closed, and at least one main steam line drain primary containment isolation valve is closed. The 12 hour Completion Time is reasonable, based on operating experience, to perform the actions from full power conditions in an orderly manner and without challenging unit systems.

An alternative to Required Actions B.1 and B.2 is to place the unit in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR, on a 31 day Frequency, requires an isotopic analysis of a representative offgas sample to ensure that the required limits are satisfied. The noble gases to be sampled are Xe-133, Xe-135, Xe-138, Kr-85m, Kr-87, and Kr-88. If the measured rate of radioactivity, as indicated by the Condenser Air Ejector Noble Gas Activity Monitor,

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

increases significantly (by ≥ 50% after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is also performed within 4 hours after the increase is noted, to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. The 31 day Frequency is adequate in view of other instrumentation that continuously monitor the offgas, and is acceptable, based on operating experience.

This SR is modified by a Note indicating that the SR is not required to be performed until 31 days after any main steam line is not isolated and the SJAE is in operation. Only in this condition can radioactive fission gases be in the Air Ejector Offgas System at significant rates.

REFERENCES

1. Letter from J. M. Pilant (NPPD) to G. E. Lear (NRC) "Failed Fuel Pin Detection Capability," dated March 2, 1978.

2. 10 CFR 100.

3. 10 CFR 50.36(c)(2)(ii).

B 3.7 PLANT SYSTEMS

B 3.7.6 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the USAR, Section X-3.0 (Ref. 1). The assumptions of the fuel handling accident are found in the USAR, Section XIV-6.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an implicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences (calculated total effective dose equivalent at the exclusion area and low population zone boundaries) are within 10 CFR 50.67 limits (Ref. 4). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.183 (Ref. 5).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core, as discussed in the USAR, Section XIV-6.1 (Ref. 6). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 7).

BASES (continued)

LCO

The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel storage pool since the potential for a release of fission products exists.

ACTIONS

A.1

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, because irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of irradiated fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable, and all water level changes are controlled by unit procedures.

BASES

REFERENCES

1. USAR, Section X-3.0.
2. USAR, Section XIV-6.4.
3. Not used.
4. 10 CFR 50.67.
5. Regulatory Guide 1.183, July 2000.
6. USAR, Section XIV-6.1.
7. 10 CFR 50.36(c)(2)(ii).

B 3.7 PLANT SYSTEMS

B 3.7.7 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 25% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without safety relief valves opening or a reactor scram. The Main Turbine Bypass System consists of three valves connected to the main steam lines between the main steam isolation valves and the turbine stop valves. Each of these three valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Turbine Digital Electro Hydraulic (DEH) Control System, as discussed in the USAR, Section VII-11.3 (Ref. 1). The bypass valves are normally closed, and the DEH Control System controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the DEH Control System controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the bypass chest, through connecting piping, to the pressure breakdown assemblies used to further reduce the steam pressure before the steam enters the condenser.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during the high energy line break analysis, as discussed in References 2 and 3, and the feedwater controller failure maximum demand transient, as discussed in Reference 4. However, the feedwater controller failure maximum demand transient defines the MCPR operating limits if one Main Turbine Bypass Valve is inoperable.

The Main Turbine Bypass System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With one Main Turbine Bypass Valve inoperable, modifications to the MCPR operating limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow this LCO to be met. The MCPR operating limits for one inoperable Main Turbine Bypass Valve are specified in the COLR. An OPERABLE Main Turbine Bypass System requires all three bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analyses (Ref. 4).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at $\geq$ 25% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the Applicable Safety Analyses transients. As discussed in the Bases for TLCO 3.2.1, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.2.2, sufficient margin to these limits exists at < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If one Main Turbine Bypass Valve is inoperable, and the MCPR operating limits for one inoperable Main Turbine Bypass Valve, as specified in the COLR, are not applied, the assumptions of the design basis transient analyses may not be met. Under such circumstances, prompt action should be taken to restore the inoperable Main Turbine Bypass Valve to OPERABLE status or adjust the MCPR operating limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

BASES

ACTIONS (continued)

B.1

If the inoperable Main Turbine Bypass Valve cannot be restored to OPERABLE status and the MCPR operating limits for one inoperable Main Turbine Bypass Valve are not applied within 2 hours, or two or more Main Turbine Bypass Valves are inoperable, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the Applicable Safety Analyses transients. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Cycling each main turbine bypass valve through at least half of one cycle of full travel (50% open) demonstrates that the valves are mechanically OPERABLE and will function when required. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 18 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Cycling open a bypass valve at slightly above 29.5 RTP may affect the RPS Turbine Stop and Control Valve functions.

SR 3.7.7.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analyses. The response time limits are specified in the COLR. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 18 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section VII-11.3.
2. Amendment 25 to the FSAR.
3. NEDC 96-006, "Estimate of Steam Tunnel's HELB," March 3, 1996.
4. USAR, Section XIV-5.8.1.
5. 10 CFR 50.36(c)(2)(ii).

BASES

LCO (continued)

Additionally, power to the critical buses is allowed to be supplied from the NSST. In this case, the SSST offsite circuit is considered OPERABLE provided the automatic transfer capability from the NSST to the SSST is OPERABLE for both of the critical buses. For the ESST to be considered OPERABLE, the automatic transfer capability from the NSST to the ESST must be OPERABLE for both critical buses (the automatic transfer capability from the NSST to the ESST is allowed to go through an intermediate step of transferring to the first offsite source, i.e., SSST).

A verification of OPERABILITY is an administrative check, by examination of appropriate plant records (logs, surveillance test records), to determine that a system, subsystem, train, component or device is not inoperable. Such verification does not preclude the demonstration (testing) of a given system, subsystem, train, component or device to determine OPERABILITY.

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 1, 2, and 3 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of abnormal operational transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 4 and 5 are covered in LCO 3.8.2, "AC Sources — Shutdown."

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable DG. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable DG and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

ACTIONS (continued)

A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the availability of the remaining offsite circuit 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 not met. However, if the second circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

A.2

Required Action A.2, which only applies if the division cannot be powered from an offsite source, is intended to provide assurance that an event with a coincident single failure of the associated DG does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related divisions (i.e., single division systems are not included). Redundant required features failures consist of inoperable features associated with a division redundant to the division that has no offsite power.

The Completion Time for Required Action A.2 is intended to allow time for the operator to evaluate and repair any discovered inoperabilities. This Completion Time also allows 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. The division has no offsite power supplying its loads; and

b. A redundant required feature on the other division is inoperable.

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

Discovering no offsite power to one 4.16 kV critical bus of the onsite Class 1E Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with any other Class 1E bus that has offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before the unit is subjected to transients associated with shutdown.

ACTIONS (continued)

The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection may have been lost for the required feature's function; however, function is not lost. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

The 4.16 kV critical bus design and loading is sufficient to allow operation to continue in Condition A for a period that should not exceed 7 days. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the plant safety systems. In this condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

The 7 day Completion Time takes into account the redundancy, capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action A.3 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 A is entered while, for instance, a DG is inoperable, and that DG is subsequently returned 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 to meet the LCO, to restore the offsite circuit. At this time, a DG could again become inoperable, the circuit 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 Completion Time must be met.

Similar to Required Action A.2, the second Completion Time of Required Action A.3 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 the LCO was initially not met, instead of at the time that Condition A was entered.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.3

associated critical bus). Because each DG is rated at 4000 kW at 0.8 power factor (pf), the required load band is ≥ 3600 kW at ≥ 0.8 pf (≥ 90% of rated load, in accordance with Regulatory Guide 1.9, Ref. 9) and less than or equal to rated load. This load band brackets the maximum expected accident loads. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

The 31 day Frequency for this Surveillance is consistent with Regulatory Guide 1.9 (Ref. 9).

Note 1 modifies this Surveillance to indicate that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized.

Note 2 modifies this Surveillance by stating that momentary transients because of changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test.

Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations.

Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank is at or above the level at which fuel oil is automatically added. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for approximately 3.9 hours of DG operation at full load.