16A.2.1 B 2.1 SAFETY LIMITS

16A.2.1.1 REACTOR CORE: B 2.1.1 DNBR; B 2.1.2 PEAK LINEAR HEAT RATE

B 2.1 SAFETY LIMITS

B 2.1.1 Reactor Core

BASES

BACKGROUND

This Bases addresses the following Reactor Core Safety Limits:

2.1.1 DNBR
2.1.2 Peak Linear Heat Rate

Safety Limits that protect the integrity of the reactor coolant pressure boundary, the fuel and fuel cladding are required to be included in the technical specifications pursuant to 10 CFR 50.36 (Ref. 1).

Specified Acceptable Fuel Design Limits (SAFDLs) are established to prevent overheating of the fuel cladding and possible cladding perforation with the release of fission products to the reactor coolant, (Ref. 3). Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is only slightly above the coolant saturation temperature, and maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21 kw/ft. which will not cause fuel centerline melting in any fuel rod. Once established, the SAFDLs become the Safety Limits used to protect the fuel and fuel cladding.

The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). DNB occurs when the local heat flux addition (BTU per unit cladding surface area per unit time) from the fuel rods to the reactor coolant causes nucleate boiling to be replaced by a steam film along regions of the cladding. At this point, there is a sharp reduction of the heat transfer coefficient which results in a large difference between the cladding surface temperature and the coolant saturation temperature. Inside the steam film, high cladding temperatures are reaches, and a cladding-water (Zircaloy-water) reaction may take place. This chemical reaction results in oxidation

(continued)

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BASES

BACKGROUND (continued)

of the fuel cladding to a mechanically weaker form. This weaker form may lose its integrity, allowing an uncontrolled release of activity to the reactor coolant.

Centerline fuel melt occurs when the local linear heat rate of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of particulate and gaseous activity to the coolant. A second consideration of centerline melting involves redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of the melting.

Because of the above factors, DNBR and Peak Linear Heat Rate have been established as Safety Limits.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I events) and Anticipated Operational Occurrences (AOOs) (Condition II events), Ref. 3. The reactor core Safety Limits are established to preclude violation of the following fuel design criteria (Ref. 2):

1. During normal operation and AOOs, there must be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience DNB. This is referred to hereafter as the 95/95 DNB criterion.

2. During normal operation and AOOs, the hot fuel pellet in the core must not experience centerline fuel melting.

Limiting Safety System Settings (LSSSs) for the low DNBR, high local power density, high logarithmic power level, low pressurizer pressure and high linear power level trips, and limiting conditions for operation on DNBR

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

and kw/ft margin are specified such that there is a high degree of confidence that the Safety Limits are not exceeded during normal operation and anticipated operational occurrences. See Section 3.2 of the LCOs for the LSSSs.

SAFETY LIMITS

SL 2.1.1 - DNBR

Correlations predict Departure from Nucleate Boiling (DNB) and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operation and design basis AOOs is limited to [1.24] CE-1 CHF correlation and is established as a Safety Limit. Additional factors such as rod bow and spacer grid size and placement will determine the LSSS ~~valve~~ value required to ensure the Safety Limit is maintained.

SL 2.1.2 - Peak Linear Heat Rate

Maintaining peak linear heat rates to ≤ 21 kw/ft ensures fuel centerline melt will not occur during normal operating conditions or design AOOs.

APPLICABILITY

The limits on DNBR and peak linear heat rate are applicable in MODES 1 and 2 since the conditions which could lead to reaching these limits only occur when the reactor is producing THERMAL POWER. Applicability in other MODES is meaningless, since the reactor is not critical or generating THERMAL POWER.

SAFETY LIMIT VIOLATION

2.2.1 Exceeding the DNBR or peak linear heat rate Safety Limits may cause immediate fuel damage due to the Zircaloy-water reaction or centerline fuel melt. Therefore, any deviation outside these limis must be corrected as quickly as possible. Placing the unit in MODE 3 ensures the potential for DNB and centerline fuel melt will not continue. Normally this will occur as the result of an automatic reactor trip from the Reactor Protection System.

(continued)

BASES

SAFETY LIMIT VIOLATION (continued)

The 1 hour allotted to reach MODE 3 is adequate to conduct a rapid, controlled shutdown while minimizing the time the reactor is operated after exceeding a Safety Limit.

REFERENCES

1. 10 CFR 50, Licensing of Production and Utilization Facilities. Part 50.36, Technical Specifications.

2. ANSI N18.2-1973, Nuclear Safety Criteria for the Design of Stationary Pressurizes Water Reactor Plants, American National Standards Institute, August 6, 1973.

3. 10 CFR 50, Appendix A, General Criteria for Nuclear Power Plants, Criterion 10.

4. CESSAR-DC Chapter 4, Reactor.

5. CESSAR-DC Chapter 15, Accident Analysis

16A.2.1.2 B 2.1.3 RCS PRESSURE

B 2.1 SAFETY LIMITS

B 2.1.3 RCS Pressure

BASES

BACKGROUND	The function of this safety limit is to prevent the RCS from exceeding its design pressure limit by an amount greater than 10%. Per Section III of the ASME code, the limiting pressure for the RCS may not exceed 2500 psia plus 250 psia for a total of 2750 psia. Four pressurizer code safety valves, each set to lift at 2500 psia ±1% to protect the RCS from overpressurization. The applicable analysis covering the structural design of the RCS is based on Section III of the ASME code. Relieving capacity rules and guidelines are provided in the code under Section NB-7000, which pertains to the RCS. Single failure of either of the pressurizer safety valves is not a criterion for operation and is so noted in the ASME code by allowing the total relieving capacity of all safety valves to be credited to overpressure protection. The consequences of exceeding 2750 psia depends on the amount of overpressure imposed on the RCS.
APPLICABLE SAFETY ANALYSIS	There are no transient or accident analyses that assume the RCS is initially at its design pressure, or that it reaches its Pressure Safety Limit (110% of design) during any Design Bases Event.
SAFETY LIMITS	SL 2.1.3 - RCS Pressure The RCS Pressure Safety Limit protects the reactor coolant pressure boundary (RCPB) integrity. Violation of this safety limit could result in uncontrolled RCPB leakage and release of fission products to the containment.

(continued)

BASES

APPLICABILITY

During operation in MODES 1 through 5, the Reactor Vessel head bolts are fully tensioned and pressures and temperatures are elevated. A slight possibility exists that under certain circumstances, a transient could occur that would cause the RCS Pressure Safety Limit to be reached or exceeded. Therefore, the LCO applies in these MODES.

During MODE 6 operation, the Reactor Vessel head bolt(s) are either not fully tensioned, or the head is removed. In addition, the temperatures and pressures are much lower than in the other MODES. Therefore, the possibility of overpressurization does not exist and the LCO does not apply.

SAFETY LIMIT VIOLATIONS

2.2.1

With the RCS pressure greater than 2750 psig in MODES 1 or 2, the pressure must be reduced to below this value. A pressure greater than 2750 psig exceeds 110% of RCS design pressure and may challenge system integrity. Reducing RCS pressure is accomplished by terminating the cause of the pressure increase, removing energy or mass from the RCS, or a combination of the above actions.

The 15-minute Completion Time is based on the importance of reducing RCS pressure to within limits and allows the operator sufficient time to perform the actions required to reduce pressure without causing an additional plant transient.

2.2.2

With RCS pressure greater than 2750 psig in MODES 3, 4, or 5, the pressure must be reduced below this value. Exceeding the Safety Limit in MODES 3, 4, or 5 is potentially more severe than exceeding the Safety Limit in MODES 1 or 2 since the reactor vessel temperature may be lower. Reducing RCS pressure is accomplished by removing mass or energy or both from the system. In MODES 3, 4, or 5, relatively small amounts of energy are being added to the system and consequently only small amount of energy or mass must be removed. The five-minute Completion Time to restore the

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

pressure to below 2750 psig is based on the importance of reducing RCS pressure to within limits and allows the operator sufficient time to perform the actions required to reduce pressure.

Reducing MODES is not required since this would require reducing temperature which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

2.2.3, 2.2.4, 2.2.5 and 2.2.6

Whenever a Safety Limit violation occurs, 10 CFR 50.36 (Ref. 2) requires the NRC to be notified as required by 10 CFR 50.72 (Ref. 3) and 10 CFR 50.73 (Ref. 4). Notification within one hour is required by 10 CFR 50.72 due to the initiation of a shutdown required by the Technical Specifications. Within 30 days, a Licensee Event Report (LER) is required by 10 CFR 50.73 due to completion of the required shutdown. Reference 3 also requires NRC approval to resume plant operation. These Required Actions ensure the NRC and appropriate plant personnel are notified in a timely manner.

REFERENCES

1. CESSAR-DC Chapter 5, Reactor Coolant System.

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineering.

3. 10 CFR 50.36, Technical Specifications.

4. 10 CFR 50.72, Immediate Notification Requirements for Operating Nuclear Power Reactors.

5. 10 CFR 50.73, Licensee Event Report System.

Additional References

6. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Plants, February 6, 1987.

16A.3 B 3.0 APPLICABILITY

16A.3.1 LIMITING CONDITIONS FOR OPERATION (LCOs) - APPLICABILITY

B 3.0 LIMITING CONDITIONS FOR OPERATION (LCOs) - APPLICABILITY

BASES

LCOs 3.0.1 - 3.0.9	LCOs 3.0.1 through 3.0.9 establish the general requirements applicable to all Specifications unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when (i.e., in which MODEs or other specified conditions) conformance to the LCO is required.

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet a Limiting Condition for Operation, the associated ACTIONS shall be met. The Completion Times of the Required Actions are applicable from the point in time an ACTIONS' Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

1. Completion of the Required Actions within the specified Completion Times constitutes compliance with a specification, and

2. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these Required Actions are not completed within the specified

Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the specification no longer applies. The second type of Required Action specifies the remedial measures that permit continued operation of the unit which is not further restricted by the Completion Time. In this case, conformance to the Required Actions provides an acceptable level of safety for continued operation.

(continued)

BASES

LCO 3.0.2 (continued)

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillance Requirements, preventive maintenance, corrective maintenance, or investigation of operational problems. It is not intended that intentional entry into shutdown Required Actions (i.e., Actions requiring a change in MODE) be made for operational convenience. This is to limit routine voluntary removal of redundant equipment from service in lieu of other alternatives that would not result in redundant equipment being inoperable. Individual specifications may include a specified Completion Time of a Surveillance Requirement when equipment is removed from service, or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this limit expires if the surveillance has not been completed.

When a MODE or other specified condition change is required to comply with Required Actions, the unit may have entered a MODE or other specified condition in which a new specification becomes applicable. In this case, the Completion Times of any new Required Actions would apply from the point in time that the new specification becomes applicable and the ACTIONS' Condition(s) are entered.

It is allowable to be in more than one ACTIONS' Condition of a given LCO at the same time. When in multiple ACTIONS' Conditions, the Required Actions of each Condition entered must be completed within the specified Completion Time.

LCO 3.0.3

LCO 3.0.3 establishes the Required Actions that must be implemented when an LCO is not met and:

1. An associated Required Action and Completion Time is not met and no other Condition applies, or
2. The Condition of the unit is not specifically addressed by the associated ACTIONS. (ADD additional wording – next page)

(continued)

BASES (continued)

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

Add this to LCO 3.0.3, 2.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

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BASES

LCO 3.0.3 (continued)

This specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience which permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions for which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 shall be consistent with the discussion of Section 1.3, Completion Times.

A unit shutdown required per LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

1. The LCO is now met.

2. A Condition exists for which the Required Actions have now been performed.

3. ACTIONS exist which do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition was initially entered and not from the time LCO 3.0.3 is exited.

(continued)

BASES

LCO 3.0.3 (continued)

The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE of operation applies. However, if a lower MODE of operation is reached in less time than allowed, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, the time allowed to reach MODE 4 is the next 11 hours, because the total time to reach MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3 and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition that LCO 3.0.3 would require the unit to be placed in. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability, unless in MODES 1, 2, 3 or 4, because the ACTIONS of individual specifications sufficiently define the remedial measures to be taken.

The exceptions to LCO 3.0.3 are provided in instances where requiring a plant shutdown in accordance with LCO 3.0.3 would not provide appropriate remedial measures for the associated condition of the unit. These exceptions are addressed in the individual specifications.

The requirement to be in MODE 4 in 13 hours is plant specific and depends on the ability to cool the pressurizer and degas.

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 precludes placing the unit in a different MODE or other specified condition when the requirements of an LCO in the MODE or other specified condition to be entered are not met and continued noncompliance with these conditions would result in a shutdown to comply with the Required Actions.

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BASES

LCO 3.0.4 (continued)

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in the 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 a MODE change. Therefore, in this case, entry into a MODE or other condition specified in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before unit startup.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability which are required to comply with ACTIONS.

Exceptions to LCO 3.0.4 are stated in the individual specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a specification.

When changing MODES or other specified conditions while in an ACTIONS Condition in compliance with LCO 3.0.4, or where an exception to LCO 3.0.4 is stated, the ACTIONS define the remedial measures that apply. Surveillance requirements do not have to be performed on the associated inoperable equipment. Therefore, a MODE change in this situation does not violate SR 3.0.1 or SR 3.0.4 for those Surveillance Requirements that do not have to be performed due to the associated inoperable equipment. However, Surveillance Requirements must be met to demonstrate OPERABILITY prior to declaring the associated equipment OPERABLE and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance of restoring equipment to service when it has been removed from service or declared inoperable to comply with ACTIONS. This Specification provides an exception to LCO 3.0.2 to allow performance of testing to:

1. Demonstrate the OPERABILITY of the equipment being returned to service, or

2. Demonstrate the OPERABILITY of other equipment.

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BASES

LCO 3.0.5 (continued)

It is necessary to allow such testing to be conducted to prove OPERABILITY and return the equipment to service. Placing inoperable equipment in service for this testing does not affect its Completion Time.

LCO 3.0.6

~~LCO 3.0.6 establishes the allowance of delaying Required Actions for up to 8 hours to perform Surveillance Requirements when systems are rendered inoperable for the performance of the Surveillance Requirements. This specification provides an exception to LCO 3.0.2 to allow performance of Surveillance Requirements required to demonstrate OPERABILITY.~~

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LCO 3.0.7

~~LCO 3.0.7 establishes which ACTIONS are applicable when support systems are inoperable, depending on whether or not they have an LCO specified in the Technical Specifications. The supported system is not required to be declared inoperable solely due to support system inoperability. Only the support system LCO's ACTIONS are required to be entered. The supported systems' ACTIONS are only required to be entered where directed to do so by the support systems' ACTIONS. This is a clarification of the definition of OPERABILITY. This clarification is necessary to establish the relationship between the support systems and the supported systems, to preclude cascading to multiple supported system ACTIONS, and to eliminate the confusion associated with entering multiple LCOs' ACTIONS. Examples of support systems with LCOs specified in the Technical Specifications include service water, diesel generators, and AC and DC distribution.~~

~~When a support system is inoperable and there is not an LCO for that support system specified in the Technical Specifications, the licensee shall evaluate the impact of the inoperability or degradation of the support system function on the OPERABILITY of the supported systems. This is because the inoperability or degradation of the support system function may or may not affect the OPERABILITY of the supported system. OPERABILITY of the supported system will depend on the intended function of the supported system and the level of support that the support system provides. Upon determination that the supported system is inoperable, the ACTIONS of its LCO shall apply.~~

Use Next Page

(continued)

BASES

LCO 3.0.5 (continued)

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.

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 unit is maintained in a safe condition are specified in the support system 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' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

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BASES

LCO 3.0.6 (continued)

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

Specification 5.£, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon failure to meet two or more LCOs concurrently, 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 train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train 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

Special tests and operations are required at various times over the unit's life to demonstrate performance characteristics, to perform maintenance activities, and to perform special evaluations. Because TS normally preclude these tests and operations, special test exceptions (STEs) allow specified requirements to be changed or suspended under controlled conditions. STEs are included in applicable sections of the Specifications. Unless otherwise specified, all other TS requirements remain unchanged and in

(continued)

BASES

LCO 3.0.7 (continued)

effect as applicable. This will ensure that all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed or suspended to perform the special test or operation will remain in effect.

The Applicability of an STE LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with STE LCOs is optional.

A special test may be performed under either the provisions of the appropriate STE LCO or the other applicable TS requirements. If it is desired to perform the special test under the provisions of the STE LCO, the requirements of the STE LCO shall be followed. This includes the SRs specified in the STE LCO.

Some of the STE LCOs require that one or more of the LCOs for normal operation be met (i.e., meeting the STE LCO requires meeting the specified normal LCOs). The Applicability, ACTIONS, and SRs of the specified normal LCOs, however, are not required to be met in order to meet the STE LCO when it is in effect. This means that, upon failure to meet a specified normal LCO, the associated ACTIONS of the STE LCO apply, in lieu of the ACTIONS of the normal LCO. Exceptions to the above do exist. There are instances when the Applicability of the specified normal LCO must be met, where its ACTIONS must be taken, where certain of its Surveillances must be performed, or where all of these requirements must be met concurrently with the requirements of the STE LCO.

Unless the SRs of the specified normal LCOs are suspended or changed by the special test, those SRs that are necessary to meet the specified normal LCOs must be met prior to performing the special test. During the conduct of the special test, those Surveillances need not be performed unless specified by the ACTIONS or SRs of the STE LCO.

ACTIONS for STE LCOs provide appropriate remedial measures upon failure to meet the STE LCO. Upon failure to meet these ACTIONS, suspend the performance of the special test and enter the ACTIONS for all LCOs that are then not met. Entry into LCO 3.0.3 may possibly be required, but this determination should not be made by considering only the failure to meet the ACTIONS of the STE LCO.

BASES

LCO 3.0.8

Special Test Exception (STE) LCOs allow specified technical specification requirements to be changed under controlled conditions. Compliance with a STE is not required at all times its applicability may be met, rather compliance is invoked at the option of the licensee.

Upon discovery of a failure to meet an LCO superseded by a STE, the ACTIONS of the STE supersede those of the LCO. In such a case, the ACTIONS of the STE shall be taken rather than those of the failed LCO. The provisions of LCO 3.0.3 are not applicable to STEs, as a unit shutdown may not be the correct action to take from the plant conditions.

When a STE is not in effect, entry into a MODE or other specified condition in its Applicability does not automatically invoke it.

LCO 3.0.9

LCO 3.0.9 establishes the applicability of each Specification to Unit 1 and Unit 2 operations (for multi-unit sites with common control rooms).

16A.3.2 SURVEILLANCE REQUIREMENTS

B 3.0 SURVEILLANCE REQUIREMENTS - APPLICABILITY

BASES

SRs 3.0.1 - 3.0.5	SRs 3.0.1 through 3.0.5 establish the general requirements applicable to all Specifications.

SR 3.0.1

SR 3.0.1 establishes the requirement that Surveillance Requirements must be met during the MODES or other specified conditions in the Applicability of the LCO, unless otherwise specified in the individual Surveillance Requirements. This Specification ensures that Surveillances are performed to verify the OPERABILITY of systems and components, and that parameters are within specified limits. Failure to meet a Surveillance Requirement 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 Surveillance Requirements have been met. However, nothing in this Specification is to be construed as implying that systems or components are OPERABLE when:

1. They are known to be inoperable although still meeting the Surveillance Requirements, or

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

Surveillance Requirements do not have to be performed when the unit is in a MODE or other specified condition for which the associated LCO is not applicable, unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is invoked.

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

(continued)

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post-maintenance testing is required to declare equipment OPERABLE. This includes meeting applicable Surveillance Requirements 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 allows operation to proceed to a MODE or other specified condition where other necessary post-maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillance, Required Actions calling for the performance of a Surveillance Requirement, and any Required Action with a Completion Time requiring the periodic performance of an action on a "once per..." interval. SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This facilitates surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the Surveillance, e.g., transient conditions or other ongoing surveillance or maintenance activities.

The 25% extension does not significantly degrade the reliability preserved by performing the Surveillance at its specified Frequency. This recognizes that the most probable result of any particular Surveillance Requirement being performed is the verification of conformance with the Surveillance Requirements. The exceptions to SR 3.0.2 are those Surveillance where the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is a Surveillance with a Frequency of "In accordance with 10CFR50 Appendix J, and approved exemptions". The requirements of regulations take precedence over the Technical Specifications. The Technical Specifications cannot, in and of themselves, extend a test interval specified in the regulations.

(continued)

BASES

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring equipment inoperable when Surveillances have not been completed within the specified Frequency. An allowed 24 hour deferral applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This deferral provides an adequate time to complete Surveillance that have been missed. This allowance permits the completion of a Surveillance before compliance with Required Actions or other remedial measures would be required that may preclude completion of a Surveillance. The basis for this allowance includes consideration for 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 Surveillance Requirements. This provision also provides a time limit for completion of Surveillance Requirements that become applicable as a consequence of MODE changes imposed by Required Actions.

If a Surveillance is not completed within the 24 hour allowance, the equipment is considered inoperable and the Completion Times of the Required Actions begin immediately upon expiration of the 24 hour allowance. If a Surveillance is failed within the 24 hour allowance, the equipment is considered inoperable and the Completion Times of the Required Actions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance Requirement in accordance with the 24 hour allowance of this Specification, or within the Completion Time of the ACTIONS restores compliance with the requirements of SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable Surveillances must be met before entry into a MODE or other specified condition in the Applicability. This Specification ensures that system and component OPERABILITY requirements, or parameter limits, are met before entry into a MODE or other specified condition in the Applicability in which the OPERABILITY requirements apply. This provision applies to changes in MODES or other specified conditions in the Applicability associated with unit shutdown, as well as startup.

(continued)

BASES

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability which are required to comply with ACTIONS.

Exceptions to LCO 3.0.4 are stated in the individual Surveillance Requirements. Exceptions to this Specification are allowed to establish the conditions required to perform surveillance testing when the prerequisite condition(s) specified in a surveillance test procedure require entry into the MODE or other specified condition for performance of the test. An exception to SR 3.0.4 automatically allows 24 hours for completion of a Surveillance Requirement from the time the prerequisite test conditions are met. If 24 hours is not sufficient to complete the Surveillance Requirement, an appropriate time interval is stated with the exception to the specific Surveillance requirement. Unless otherwise known to be inoperable, equipment or systems are not considered inoperable and ACTIONS Conditions are not entered while complying with an exception to SR 3.0.4. The associated Required Actions become applicable only if the Surveillance Requirement(s) is not met within the time allowed by the exception.

SR 3.0.5

SR 3.0.5 establishes the applicability of the Surveillance activities to Unit 1 and Unit 2 operations (for multi-unit sites with common control rooms).

REFERENCES

1. NRC Generic Letter 89-14, re elimination of 3.25 restriction on Surveillance Frequency.

2. "Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operation and Surveillance Requirements," NRC Generic Letter 87-09, June 4, 1987.

16A.4 B 3.1 REACTIVITY CONTROL SYSTEMS

16A.4.1 B 3.1.1 SHUTDOWN MARGIN ~~- T_{AVG} > 210°F~~ 138°F

135°F

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 Shutdown Margin - ~~T_{avg} > 210°F~~ 135°F

BASES

BACKGROUND

This Bases addresses SHUTDOWN MARGIN (SDM) requirements when the reactor is not generating THERMAL POWER. ~~and T_{avg} is > 210°F.~~

The reactivity control system must be redundant and capable of holding the reactor core subcritical when shutdown under cold conditions (General Design Criterion 26, Ref. 1). Maintenance of the SDM ensures that postulated reactivity events will ~~not damage the fuel~~ have acceptable as defined by the SRP (Ref 4).

Two independent reactivity control systems are provided, soluble boron and control element assemblies (CEAs). The CEAs are used to shut down the reactor from normal operation and during anticipated operational occurrences (AOOs) and accidents. During power operation, SDM control is verified by operating the core within the Regulating CEA insertion limits of LCO 3.1.7 with the Shutdown CEAs fully withdrawn, (LCO 3.1.6). This ensures that the amount of reactivity required to shutdown the reactor to MODE 3 conditions is immediately available upon a reactor trip at any time in core life.

After a reactor shutdown or trip, the CEAs should be fully inserted. Adjustments to the Reactor Coolant System (RCS) boron concentration are then used to maintain the required SDM as RCS temperature decreases and as xenon decays.

Maintenance of the required SDM in MODES 2 subcritical, 3 ~~and~~ 4 and 5 is necessary to ensure that ~~accidents~~ events such as ~~a startup accident, ejected CEA, steam system piping failure, or moderator dilution event initiated from a shutdown condition~~ will not result in unacceptable consequences. ~~The SDM requirement is not however, based on a MODE 3 or 4 transient.~~ The value of the minimum SDM requirement is based on a main steam line break (MSLB) at end of core (EOC) as described in the Applicable Safety Analysis section.

1. an uncontrolled CEA withdrawal from subcritical conditions,
2. CEA ejection,
3. inadvertent deboration,
4. startup of an inactive RCP, or
5. steam system piping failures

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The value of the SDM ~~requirements~~ of LCO 3.1.1 ~~are~~ is based on a MSLB as described in the Accident Analysis, (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator, and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. Negative reactivity is added initially by the control rods and then by boron injected by the SI pumps. As RCS temperature decreases, the severity of a MSLB decreases until the MODE 5 value is reached. In practice, this calculation is not made, and the requirement is kept at the maximum value for MODE 3 throughout MODES 3 and 4.

The most limiting MSLB event with respect to defining the shutdown margin requirement is a guillotine break of a main steam line initiated at the end of core life at no load conditions coincident with a loss of off-site power. For this case, no return to power is predicted by the safety analyses. ~~This ensures that fuel clad integrity is maintained.~~ Consequently, there is no approach to unacceptable off-site radiation releases or to loss of a coolable core geometry.

The most limiting MSLB, with respect to return to power, is a guillotine break of a main steam line inside containment initiated at the end of core life at full power with off-site power available.

In addition to the limiting MSLB transient, the SDM SAFETY requirement must also protect against:

(1) an uncontrolled CEA withdrawal from a subcritical or low power condition,
(2) CEA ejection,
(3) inadvertent boron dilution, and
(4) startup of an inactive Reactor Coolant Pump (RCP).

Each of these events is discussed below.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The withdrawal or ejection of CEA's from subcritical or low power conditions adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The withdrawal of CEAs also produces a time dependent redistribution of core power.

In Modes 2 subcritical, 3 ~~and~~ 4, and 5, depending on the system initial conditions and reactivity insertion rate, the uncontrolled CEA withdrawal transient is terminated by either low DNBR trip, a high local power density trip or a high logarithmic power level trip. ~~In all cases,~~ Power level, RCS pressure, and the departure from nucleate boiling ratio (DNBR) do not exceed allowable limits for the inadvertent withdrawal of a regulating CEA bank (Ref 5). The LCO (see below) for this Technical Specification ensures that the reactor will not become critical due to the inadvertent withdrawal of a shutdown CEA group.

INSERT B

The limiting boron dilution event occurs when the plant is in MODE 5. ~~when SDM requirements are reduced. BASES 3.1.2 provides a discussion of the SDM requirements for MODE 5.~~

Insert C

The startup of an inactive RCP will not result in consequences more adverse than those for the CEA withdrawal event initiated from subcritical conditions.

The SDM satisfies the requirements of Criterion 2 of the Interim Policy Statement as described in Reference 3.

LCO

The accident analysis has shown that the required SDM is sufficient to avoid unacceptable consequences to the fuel or RCS as a result of the events addressed above. The zero power MSLB establishes this value. Shutdown boron concentration requirements assume the highest worth CEA is stuck in the fully withdrawn position to account for a postulated inoperable or untrippable CEA prior to reactor shutdown.

INSERT D

The LCO has been modified by a Note which states the CEA of highest reactivity worth does not have to be assumed withdrawn when all CEAs are verified inserted by two diverse position indicators. This means the worth of the most reactive CEA does not have to be included in the calculation of SDM thus eliminating unnecessary boration and dilution.

(continued)

Insert B

The LCO (see below) for this Technical Specification ensures that a CEA ejection event postulated to be initiated ~~[illegible]~~ in MODE 2 subcritical or MODEs 3, 4, or 5 will have less adverse consequences than the event analyzed for a CEA ejection in MODE 1, which has been shown to have acceptable consequences (Ref. 2).

Insert C

ment In this transient SDM is lost as the boron concentration of the reactor coolant is diluted by the addition of unborated makeup water.

For a limiting inadvertent deboration event to occur, a charging pump must be running, a primary makeup pump must be running and the demineralized water supply system must be aligned to supply water directly to the charging pump suction.

Since at least three simultaneous equipment malfunctions would be required to produce these conditions, the incident could only be the result of operator action accompanied by a single equipment malfunction. Should this event occur, the high neutron flux alarm on the startup channels or the high reactor makeup water flow alarm will alert the operator at least 30 minutes prior to reaching criticality. The event can then be terminated by either:

(1) turning off the charging pumps,
(2) turning off the primary makeup pump,
(3) isolate the reactor makeup water supply,
(4) isolate the volume control tank, or
(5) actuate safety injection.

An inadvertent boron dilution is a moderate frequency incident as defined in Chapter 15 of CESSAR-DC (Ref. 2). The core is initially subcritical with all but the highest worth CEA inserted. A Chemical and Volume Control System (CVCS) malfunction occurs which causes unborated water to be pumped to the RCS via single charging pump.

During the event coolant will be circulated through the RCS by the shutdown cooling system: complete mixing of boron within the RCS is assumed. A cold (210°F) RCS volume, of [4,400] ft^3 is assumed. This RCS volume excludes the pressurizer, surge line, reactor vessel above the midplane of the hot legs, and portions of the hot and cold legs, volumes. A dilution rate of [180] gallons per minute results in a [25] lb/sec dilution flow rate. At Beginning of Core (BOC) conditions the initial boron concentration is [1012] ppm with all CEAs inserted except for the worth of the highest worth CEA. The inverse boron worth assumed in this condition is [66] ppm/% $\Delta k/k$.

Using beginning of core conservative parameters results in a minimum possible time interval to dilute to criticality of approximately [38] minutes. The high reactor makeup water flow alarm will alert the operator of the event. Boron dilution can then be terminated by operator action.

INSERT D

LCO 3.1.1 b.1 requires that the calculated critical position be within the limits of Technical Specifications 3.1.6 and 3.1.7 when the reactor trip breakers are closed. This ensures that the most adverse subcritical CEA withdrawal event scenario is the inadvertent withdrawal of a regulating CEA bank, i.e. the reactor will not become critical due to the inadvertent withdrawal of a shutdown CEA bank.

LCO 3.1.1 b.1 also ensures that, if the RTCBs are closed, a CEA ejection event postulated to be initiated at these conditions would result in less net positive reactivity insertion than for a case initiated from a critical position. If the RTCBs are open, LCO 3.1.1 b.2 requires that the value of k_{eff} must remain less than 1.0 when the highest worth CEA is excluded from the calculation. This, latter, requirement ensures that the reactor would not reach criticality for a CEA ejection event posculated to be initiated under these conditions. Together, therefore, LCO 3.1.1 b.1 and LCO 3.1.1 b.2 ensure that a CEA ejection event postulated to be initiated in MODE 2 subcritical or MODEs 3, 4, or 5 would have less adverse consequences than the event analyzed for a CEA ejection in MODE 1, which has been shown to have acceptable consequences (Ref. 2).

BASES

APPLICABILITY

2 subcritical, and 5.
In MODES 3 ~~and 4~~, sufficient negative reactivity is required to protect against the reactivity addition events discussed. SDM in MODES 1 and 2 is addressed in LCO 3.1.6, Shutdown CEA Insertion Limits, and LCO 3.1.7, Regulating CEA Insertion Limits.

~~SDM in MODE 5 is addressed in LCO 3.1.2, SHUTDOWN MARGIN - T_{avg} ≤ 210°F.~~

SDM in MODE 6 is addressed in LCO 3.9.1, Boron Concentration.

ACTIONS

A.1

Restoration of the SDM requires increasing the RCS boron concentration. The required Completion Time to initiate boration allows the operator time to align the required valves and start the required pumps. Boration should continue until the SDM is restored.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

The SDM is verified by performing a reactivity balance calculation. Reactivity contributions due to boron concentration, CEA position, moderator temperature, fuel depletion, xenon concentrations, and samarium concentration are summed and compared to the SDM requirement. Doppler reactivity is excluded from the reactivity balance since the reactor is at zero power in all applicable MODES and the effects of fuel temperature changes are negligible.

The required frequency of 24 hours is based on the low probability of an accident occurring, industry-accepted practice, and the steps required to perform the surveillance. This allows the operator time to collect the data required, including results from analysis of a RCS boron sample, and perform the reactivity balance calculation. This frequency has been shown to be acceptable through operating experience.

2 subcritical, or 5
If CEA is inoperable, but ~~in the~~ tripped position (fully inserted) with the reactor in MODE 3 ~~or 4~~ then the SDM requirement need not be increased by the worth of the inoperable CEA.

(continued)

BASES

REFERENCES

1. 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, GDC 26, Reactivity Control System Redundancy and Capability.

2. System 80+ CESSAR-DC, Section 15.1.3

3. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application)."

4. NUREG-0800 USNRC Standard Review Plan

~~5. DETR. 10~~

5. CESSAR DC, Section 19.8.

135°F

16A.4.2 B 3.1.2 ~~SHUTDOWN MARGIN - T_{avg} ≤ 210°F~~ (DELETED)

135°F

~~Shutdown Margin - T_{avg} ≤ 210°F~~
B 3.1.2

B 3.1 REACTIVITY CONTROL SYSTEMS

135°F

B 3.1.2 ~~Shutdown Margin - T_{avg} ≤ 210°F~~ (Deleted)

BASES

BACKGROUND

135°F

This Bases addresses maintaining SDM requirements when T_{avg} is ≤ ~~210°F~~.

135°F

With T_{avg} ≤ ~~210°F~~ reactivity transients resulting from a postulated accident involving a cooldown are minimal. This is because of the reduced rate of change in Reactor Coolant System (RCS) temperature which can occur and the reduced value of Moderator Temperature Coefficient (MTC) in this temperature range.

The shutdown margin requirement must protect against:

1. an uncontrolled CEA withdrawal from subcritical conditions
2. CEA Ejection
3. Inadvertent deboration
4. Startup of an inactive Reactor Coolant Pump (RCP)

The CEA Ejection event in Modes 5 or 6 can be considered incredible due to the low RCS temperatures experienced in these modes. The CEA withdrawal event is terminated by action of reactor protection system. The inadvertent startup of an inactive RCP has been shown in Chapter 15 of CESSAR-DC to result in acceptable consequences. Thus, the limiting SDM transient in this condition is therefore, an inadvertent deboration of the RCS. In this transient SDM is lost as the boron concentration of the reactor coolant is diluted by the addition of unborated makeup water.

For a limiting inadvertent deboration event to occur, a charging pump must be running, a primary makeup pump must be running and the demineralized water supply system must be aligned to supply water directly to the charging pump suction.

(continued)

BASES

BACKGROUND (continued)

Since at least three simultaneous equipment malfunctions would be required to produce these conditions the incident could only be the result of operator action accompanied by a single equipment malfunction. Should this event occur, the high neutron flux alarm on the startup channels or the high reactor makeup water flow alarm will alert the operator at least 30 minutes prior to reaching criticality. The event can then be terminated by either:

(1) turning off the charging pumps,
(2) turning off the primary makeup pump,
(3) isolate the reactor makeup water supply,
(4) isolate the volume control tank, or
(5) actuate safety injection.

The reactivity control system must be redundant and capable of holding the reactor core subcritical when shutdown under cold conditions (General Design Criterion 26, Ref. 1). Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

APPLICABLE SAFETY ANALYSES

An inadvertent boron dilution is a moderate frequency incident as defined in Chapter 15 of CESSAR-DC (Ref. 2). The core is initially subcritical with all but the highest worth CEA inserted. A Chemical and Volume Control System (CVCS) malfunction occurs which causes unborated water to be pumped to the RCS via single charging pump.

During the event coolant will be circulated through the RCS by the shutdown cooling system: complete mixing of boron within the RCS is assumed. A cold (210°F) RCS volume, of [4,400] ft^3 is assumed. This RCS volume excludes the pressurizer, surge line, reactor vessel above the midplane of the hot legs, and portions of the hot and cold legs, volumes. A dilution rate of [180] gallons per minute results in a [25] lb/sec dilution flow rate. At Beginning of Core (BOC) conditions the initial boron concentration is [1012] ppm with all CEAs inserted except for the worth of the highest worth CEA. The inverse boron worth assumed in this condition is [66] ppm/% Δk/k.

(continued)

Amendment I
December 21, 1990

BASES

APPLICABLE SAFETY ANALYSES (continued)	Using beginning of core conservative parameters results in a minimum possible time interval to dilute to criticality of approximately [38] minutes. The high reactor makeup water flow alarm will alert the operator of the event. Boron dilution can then be terminated by operator action. SDM satisfies the requirements of Criterion 2 of the Interim Policy Statement as described in Reference 3.
LCO	The accident analysis (Ref. 2) has shown that the required SDM is sufficient to avoid unacceptable consequences to the fuel or RCS as a result of the accidents addressed above.
APPLICABILITY	The required SDM is sufficient to control accidents and anticipated operational occurrences that are postulated to occur in MODE 5. SDM in MODES 1 and 2 is addressed by LCO 3.1.6, Shutdown CEA Insertion Limits, and LCO 3.1.7, Regulating CEA Insertion Limits. SDM in MODES 3 and 4 is addressed in LCO 3.1.1, SDM - T_{avg} > 210°F. SDM in MODE 6 is addressed by LCO 3.9.1, Boron Concentration.
ACTIONS	A.1 Restoration of the SDM requires increasing the RCS boron concentration. The required Completion Time to initiate boration allows the operator sufficient time to align the required valves and start the changing or boric acid pumps.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

The SDM is verified by performing a reactivity balance calculation. Reactivity changes due to boron concentration, CEA position, moderator temperature, fuel depletion, xenon concentrations, and samarium concentration are summed and compared to the SDM requirement. Doppler reactivity is excluded from the reactivity balance since the reactor is at zero power in all applicable MODES and there will be no fuel temperature changes.

The required frequency of 24 hours allows the operator time to collect the data required, including results from analysis of a RCS boron sample, and perform the reactivity balance calculation. This frequency has been shown to be acceptable through operating experience.

REFERENCES

1. 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, GDC 26, Reactivity Control System Redundancy and Capability.

2. System 80+ CESSAR-DC, Chapter 15

3. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application)."

16A.4.3 B 3.1.3 REACTIVITY BALANCE

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Reactivity Balance

BASES

BACKGROUND

This Bases addresses core reactivity balance when the reactor is critical or operating at power.

When the reactor is critical or in normal power operations, a reactivity balance exists. The positive reactivity of the fuel is balanced by the negative reactivity of the control components, thermal feedback, and materials in the core that absorb neutrons. Reactivity is measured by the critical boron curve, which provides an indication of the soluble boron concentration requirement in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value provides a convenient method of ensuring that core reactivity is within design expectations, and that the calculational models used to generate the predictions are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading provides excess positive reactivity beyond that required to sustain steady-state operation at the beginning of cycle (BOC). When the reactor is critical at RATED THERMAL POWER and average moderator temperature, the excess positive reactivity is compensated by the burnable poison rods (if any), regulating Control Element Assemblies (CEAs), whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

When the core is producing THERMAL POWER, the fuel is being depleted of positive reactivity. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The critical boron curve is based on operation in steady-state at RATED THERMAL POWER. Therefore, deviations from the predicted critical boron curve may indicate deficiencies in the design analysis, the calculational models, or in core conditions, and must be evaluated.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS

The NRC regulations and General Design Criteria do not address the long-term core reactivity balance directly. However, General Design Criterion 26 (Ref. 1) requires that the reactivity control system be capable of reliably controlling reactivity changes during normal operation. Regulation of the RCS boron concentration by the chemical volume and control system provides long-term control of reactivity due to fuel depletion.

Safety analyses are not performed in the Final Safety Analysis Report to address core reactivity during long term fuel depletion. Design calculations are performed for each fuel cycle that form the basis of determining the RCS boron concentration requirements for control of the reactivity balance during fuel depletion. This Specification is provided to ensure that core reactivity behaves as expected in the long term, and to ensure the NRC is notified if significant reactivity anomalies develop during reactor operation.

The comparison between measured and predicted core reactivity provides a benchmark for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at BOC do not agree, then the design analysis may contain errors, or the calculational models used to predict soluble boron requirements may not be adequate. If reasonable agreement between measured and predicted core reactivity at BOC exists, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted critical boron curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RATED THERMAL POWER following startup, with the CEAs in their normal positions for power operation. The normalization is performed at BOC conditions so that core reactivity can be continually monitored and evaluated as core conditions change thereafter.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS

When a reactivity balance exists, the core reactivity parameters sum to zero reactivity, indicating the core has reached steady-state conditions for a given THERMAL POWER level and fuel burnup. Perturbations to core reactivity may be introduced by normal power level maneuvering, reactor shutdown and startup, xenon and samarium transients, fuel depletion, changes in RCS average temperature, CEA movements, and RCS boron concentration changes. Therefore, evaluations of differences in measured and predicted core reactivity must include consideration of these parameters to determine that core conditions during the boron concentration measurement are consistent with those input to the calculational models for prediction.

The NRC specified that the core reactivity anomaly specification satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors (Ref. 2), and that this LCO shall be retained in the Technical Specifications.

LCOs

When measured core reactivity is within 1% $\Delta k/k$ of the predicted value at steady-state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady-state RCS critical boron concentrations, the difference between measured and predicted values would be approximately [130 ppm to 100 ppm] (depending on the inverse boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODE 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER, and because the fuel is depleting during power operation. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This specification does not apply in MODES 3, 4, 5 or 6 because the reactor is shut down with a minimum SDM, is not producing THERMAL POWER, and the fuel is not being depleted.

(continued)

BASES

ACTIONS

A.1

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety evaluation is performed. In practice, smaller deviations in core reactivity (less than 0.5% $\Delta k/k$) are generally cause for concern, and evaluation of both core conditions and the core design are performed to determine the cause of the deviation.

When a reactivity deviation is noted, the evaluation of core conditions typically includes the following steps:

a. Core conditions comprising the input to calculational models are verified to exist.

b. Shutdown capability from both the CEAs and the boron injection system is determined to be adequate.

c. A core power distribution map is obtained to evaluate peaking factors.

d. OPERABILITY of all CEAs is verified.

An evaluation of the core design and safety analysis typically includes the following steps:

a. Reactivity worth calculations of boron, the CEAs, xenon, and samarium are reviewed.

b. The moderator temperature coefficient calculation is reviewed and verified to be within the bounds of the safety analysis.

c. The fuel depletion calculations are reviewed to determine that the calculated core burnup is appropriate.

d. The calculational models are reviewed to verify that they are adequate for representing the core conditions.

e. The fuel temperature coefficient calculation is reviewed and verified to be within the bounds of the safety analysis.

(continued)

BASES

ACTIONS (continued)

Reactivity anomalies are generally investigated when they are small, so that the evaluations are in progress before the 1% $\Delta k/k$ reactivity limit for a deviation is reached, and corrective measures may be defined. The required Completion Time of 72 hours is adequate to prepare final documentation of the evaluation and to prepare whatever operating restrictions or surveillances that may be required to allow continued reactor operation, per Required Action A.2.

A.2

Following evaluations of the core design and safety evaluation, the cause of the reactivity anomaly may be explained. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If the cause of the reactivity anomaly is an error in the predictions, then the predictions may be recalculated to eliminate the error. If the cause of the reactivity anomaly is in the calculational technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are acceptable for continued operation, then the critical boron curve may be renormalized, and power operation may continue. If operational restrictions or additional surveillance requirements are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 72 hours is adequate to prepare whatever operating restrictions or surveillances that may be required to allow continued reactor operation.

B.1

The unit must be placed in a MODE in which the LCO does not apply if the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit by the methods discussed in Required Actions A.1 and A.2 and their associated Completion Times. This is done by placing the unit in at least MODE 3 in six hours. The allowed Completion Time is reasonable based on operating experience to reach the required MODE from RATED THERMAL POWER without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

Core reactivity is verified by periodic comparison of measured and predicted RCS boron concentrations. The comparison is made considering core conditions that include CEA position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The required frequency of 31 EFPD after entering MODE 1 is acceptable based on engineering judgment, the low probability of an accident occurring, the slow rate of core changes due to fuel depletion, and industry-accepted practice. This allows the operator to collect the data required, including results from analysis of a RCS boron concentration sample, and to perform the reactivity balance calculation. The surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The normalization of predicted core reactivity to the measured value must take place within the first 60 EFPD after each fuel loading. This allows sufficient time for core conditions to reach steady-state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations.

REFERENCES

1. Title 10 Code of Federal Regulations (10 CFR), Part 50, Appendix A, General Design Criterion 26, Reactivity Control System Redundancy and Capability.

2. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 3, 1987.

16A.4.4 B 3.1.4 MODERATOR TEMPERATURE COEFFICIENT

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Moderator Temperature Coefficient

BASES

BACKGROUND

The reactor core and its interaction with the Reactor Coolant System (RCS) are designed for stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must tend to compensate for reactivity increases.

The Moderator Temperature Coefficient (MTC) relates a change in neutron multiplication to the change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity increases with decreasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will be self-limiting and stable power operation will result. The same characteristic is true when the MTC is positive and coolant temperature decreases power.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements.

The actual value of the MTC is dependent on characteristics such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional distributed poisons (burnable poison assemblies) to yield an MTC within the range analyzed in the accident analysis. The End of Cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles designed to achieve high burnups or with changes to other characteristics, are evaluated to ensure the MTC does not exceed the EOC limit.

APPLICABLE SAFETY ANALYSIS

The System 80+ CESSAR Design Certification (CESSAR-DC), (Ref.1), contains analyses of accidents that result in both increased and decreased heat removal events. The MTC is one of the controlling parameters in many of these accidents. Therefore, both the most positive

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the CESSAR-DC safety analyses consider worst case conditions, such as high boron concentrations, to ensure the accident results are bounding.

Decreased heat removal events must be evaluated for results when the MTC is positive. Such accidents include the rod withdrawal transient from either zero or full THERMAL POWER, loss of condenser vacuum, loss of main feedwater flow, and loss of forced reactor coolant flow.

Increased heat removal events must be evaluated for results when the MTC is negative. Such accidents include sudden feedwater flow increase, sudden decrease in feedwater temperature, and steam line break events. For the EOC steam line break event, even though the reactor has tripped, the large moderator temperature reduction combined with the large negative MTC may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, reactor restart may result with all CEAs inserted except the most reactive one. However, the safety analyses shows no return to power for a MSLB event. This transient is discussed in the Applicable Safety Analysis section for LCO 3.1.1, SDM ~~$T_{avg} > 210°F$~~.

LCO 3.1.4 provides limits on MTC to ensure the core operates within the assumptions of the accident analysis. At the core design stage, the core is analyzed to predict its MTC limits over the cycle life. Variations in the MTC due to fuel depletion and boron concentration are expected to occur slowly over a period of tens of effective full power days (EFPD). The limit on positive MTC assures that core overheating accidents will not violate the accident analysis assumptions. The limit on negative MTC ensures that core overcooling accidents will not violate the accident analysis assumptions.

In Reference 3, the NRC specified that MTC meets Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 4), and that the LCO be retained in Technical Specifications.

LCO

LCO 3.1.4 requires the MTC to be within specified limits to ensure the core operates within the assumptions of the accident analysis.

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BASES

APPLICABILITY

The limits on MTC must be maintained in MODE 1 to assure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. The limits must also be maintained in MODE 2 to ensure startup and subcritical accidents (such as the uncontrolled CEA or group withdrawal) will not violate the assumptions of the accident analysis. Applicability for other MODES is not required, since neither power distribution, reactivity, or burnup-related criteria would be exceeded.

ACTIONS

A.1

MTC is a function of the fuel and fuel cycle design and cannot be controlled directly once their designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3 with a sufficiently safe SDM. This eliminates the potential for violation of the accident analysis bounds. The required Completion Time of six hours is reasonable, based on operating experience to reach MODE 3 from full power without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

The Surveillance Requirements for measurement of the MTC at the beginning and middle of each fuel cycle provide for confirmation of the limiting MTC values. The MTC changes smoothly from most positive (least negative) to most negative value during fuel cycle operation as the RCS boron concentration is reduced to compensate for fuel depletion. The requirement for measurement prior to operation above 5% of RTP satisfies the confirmatory check on most positive (least negative) MTC value. The requirement for measurement within seven days after reaching 40 EFPD and 2/3 core burnup satisfies the confirmatory check of the most negative MTC value. The measurement is performed at any THERMAL POWER so that the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. MTC values may be extrapolated and compensated to permit direct comparison to the specified MTC limits.

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BASES

SURVEILLANCE REQUIREMENTS (continued)	The SR is modified by a Note which states SR 3.0.4 is not applicable. Although this surveillance is applicable in MODE 2, the reactor must be critical before the surveillance can be completed. Therefore, entry into the Applicable MODE prior to accomplishing the surveillance is necessary.
REFERENCES	1. System 80+ CESSAR-DC, Chapter 15, Accident Analysis, Sections 15.1 and 15.2. 2. Core Operating Limits Report, to be developed. 3. "NRC Staff Review of Nuclear Steam Supply Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," transmitted by Thomas E. Murley (NRC) letter to Joseph K. Gasper (CEOG) dated May 9, 1988. 4. 52 FR 3788, NRC Interim Policy Statement, on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987. 5. 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, GDC 11, Reactor Inherent Protection.

16A.4.5 B 3.1.5 CONTROL ELEMENT ASSEMBLY (CEA) ALIGNMENT AND INSERTION LIMITS

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Element Assembly (CEA) Alignment and Insertion Limits

BASES

BACKGROUND

This Bases addresses the Limiting Conditions for Operation (LCOs) and Surveillance Requirements for the following LCOs:

3.1.5 CEA Alignment
3.1.6 Shutdown CEA Insertion Limits
3.1.7 Regulating CEA Insertion Limits

These LCOs are required to ensure proper SDM, maintain acceptable power distribution limits and limit the potential effects of CEA misalignments to acceptable levels. Since the limits protected by these LCOs share common requirements, their Bases are combined.

The Control Element Assemblies (CEAs) and Part-Strength CEAs are arranged into groups that are radially symmetric. Therefore, movement of the CEAs does not introduce radial asymmetries in the core power distribution. The Shutdown and Regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The Regulating CEAs also provide reactivity (power level) control during normal operation and transients. Their movement may be automatically controlled by the Reactor Regulating System. The Part-Strength CEAs are used to control the axial power distribution and for daily load follow.

The establishment of Limiting Safety System Settings (LSSS) and LCOs require that the expected long and short term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed based on the expected mode of operation of the Nuclear Steam Supply System (NSSS) (base loaded, maneuvering, etc.). From these analyses CEA

(continued)

BASES

BACKGROUND (continued)

insertions are determined and a consistent set of radial peaking factors defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The Long and Short Term Insertion Limits of LCO 3.1.7 are specified for the plant which has been designed for primarily base loaded operation but which also has the ability to accommodate load maneuvering.

Mechanical or electrical failures may cause a CEA to become misaligned from its group. CEA misalignment may cause a reduction in the total available rod worth for reactor shutdown and increased power peaking due to the asymmetric reactivity distribution. Therefore, CEA alignment is related to the core design requirement for minimum SDM and operation within design power peaking limits. The applicable criteria for these design requirements are 10 CFR 50, Appendix A, General Design Criteria 10 (Reactor Design), 26 (Reactivity Limits)(Ref. 2), and 10 CFR Part 50.46 Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants, (Ref. 1).

Limits on CEA alignment and safety rod position have been established, and the CEA positions are monitored and controlled during power operation to ensure that the power distribution and peaking limits defined by the power peaking and SDM limits are preserved. LCOs 3.2.5 (ASI), 3.2.3 (T_q), and 3.1.7 (Regulating CEA Insertion Limits) provide limits on monitored process variables and normal operating controls that assure steady state and transient operation is maintained within core operating limits. These LCOs assume the core is operated within the CEA alignment and position limits of LCOs 3.1.6 and 3.1.7.

The consequences of operating the reactor in violation of the Regulating CEA alignment limits is potential power peaking in excess of the design limits. The consequence of operation in violation of the Power Dependent Insertion Limit (Transient Insertion Limit) is a potential SDM below the required minimum. Therefore, the initial condition criteria for the accidents analysis would be violated for operation outside the limits.

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BASES

APPLICABLE SAFETY ANALYSES

Operation outside the CEA position and alignment LCOs could result from misoperation of the CEAs. The Accident Analysis, (Ref. 3), defines CEA misoperation as any event, with the exception of sequential group withdrawals, which could result from a single malfunction in the reactivity control system. For example, CEA misalignment may be caused by a malfunction of the Control Element Drive Mechanism (CEDM), CEDM Control System (CEDMCS), or by operator error. A stuck CEA may be caused by mechanical jamming of the CEA fingers or of the gripper. Inadvertent withdrawal of a single CEA may be caused by the opening of the electrical circuit of the CEDM holding coil for a full strength or part strength CEA. A dropped CEA subgroup could be caused by an electrical failure in the CEA coil power programmers.

The effect of any misoperated CEA on the core power distribution will be assessed by the CEA calculators (CEACs), and an appropriately augmented power distribution penalty factor will be supplied as input to the Core Protection Calculators. As the reactor core responds to the reactivity changes caused by the misoperated CEA and the ensuing reactor coolant and Doppler feedback effects, the CPCs will initiate a low Departure from Nucleate Boiling Ratio (DNBR) or high local power density trip signal if Specified Acceptable Fuel Design Limits (SAFDLs) are approached.

Since the CEA drop incidents result in the most rapid approach to SAFDLs caused by a CEA misoperation, the accident analysis analyzed a single full-strength CEA drop, a full-strength CEA subgroup drop, a single part-strength CEA drop and a part-strength CEA subgroup drop. The most rapid approach to the DNBR SAFDL is caused by either a single-full strength CEA drop or a part-strength CEA drop.

In the case of the full-strength CEA drop and a part-strength CEA drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which when conservatively coupled, result in local power and heat flux increases, and a decrease in DNBR. For plant operation within the DNBR and Local Power Density (LPD) LCOs, DNBR and LPD trips can normally be avoided on a dropped CEA.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

For a part-strength CEA (PSCEA) subgroup drop and a full-strength CEA subgroup drop, a distortion in power distribution, primarily axial, and a rapid decrease in core power are initially produced. If the drop of full-strength CEA subgroup is identified as not being a reactor power cutback by CEAC's then an appropriate power distribution penalty factor is applied in the CPC's and a reactor trip on low DNBR may be generated. As the dropped part-strength CEA subgroup is detected, an appropriate power distribution penalty factor is supplied to the CPCs and a reactor trip signal on low DNBR may be generated.

The results of the CEA misoperation analysis show that during the most limiting misoperation events no violations of the SAFDLs on DNBR, Peak Linear Heat Rate or Reactor Coolant System (RCS) pressure occur.

The CEA Alignment and Insertion Limits satisfy the requirements of Criterions 2 and 3 of the Interim Policy Statement as described in Reference 4.

LCO

The limits on CEA position and insertion, together with ASI and T_q assure the reactor will operate within the fuel design criteria. The required actions of these LCOs assure that deviations from the alignment limits will either be corrected or THERMAL POWER will be adjusted so that excessive local heat rates will not occur, and that the requirements on SDM and ejected rod worth are preserved.

During MODES 1 and 2 SDM is assured by maintaining the Shutdown CEAs fully withdrawn and the Regulating CEAs above the Transient Insertion Limit of LCO 3.1.7. With an inoperable CEA the SDM may be verified by performing a reactivity balance calculation. Reactivity changes due to boron concentration, CEA position, moderator temperature, fuel temperature, fuel depletion, xenon concentration and samarium concentration are summed and compared to the SDM requirement.

Operation of the core is limited to one inoperable or stuck CEA, since the CEA insertion limits that maintain the SDM are defined for the highest worth stuck CEA. If more than one CEA is inoperable, the reactor must be shut down to find the cause of the problem and correct it.

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LCO (continued)

The limit for individual CEA misalignment is [seven] inches between the highest and lowest CEAs in a subgroup. Maintaining CEA alignment within this span ensures a radially symmetric core power distribution.

APPLICABILITY

The CEA alignment and insertion limit LCOs 3.1.5, 3.1.6, and 3.1.7 are applicable in MODES 1 and 2. The operating controls and monitored process variables that control core power distribution within design limits in MODES 1 and 2 are given in LCOs 3.2.1 (Linear Heat Rate), 3.2.2 (Planar Radial Peaking Factors) and 3.2.3 (T_q). The limiting values for these variables are specified in the respective LCO's. The power distribution LCOs assume the requirements of the CEA position and insertion limits are satisfied.

In MODES 3, 4, 5 or 6, the CEA alignment and insertion LCOs do not apply because the reactor is shutdown and excessive local linear heat rates cannot occur from CEA misalignment.

ACTIONS

LCO 3.1.5 - CEA Alignment

A.1, A.2.1, and A.2.2

A CEA is inoperable if it will not move in response to signals from the CEDMCS. A CEA may become inoperable yet remain trippable. In this condition the CEA can still perform its required function of adding negative reactivity should a reactor trip be necessary. If a CEA is inoperable but trippable, continued operation in MODES 1 and 2 may continue provided the position of the inoperable CEA does not result in unacceptable peaking factors. The one hour Completion Time ensures an acceptable CEA alignment is established before xenon redistribution can generate unacceptable peaking factors.

B.1, B.2.1, B.2.2.1 and B.2.2

Xenon redistribution in the core starts to occur as soon as a CEA becomes misaligned. Reducing THERMAL POWER in accordance with Figure 3.1.5-1 ensures acceptable power distributions are maintained. When a misaligned

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BASES

ACTIONS (continued)

CEA occurs, it can usually be moved and is still trippable. For small misalignments (less than [19] inches) of the CEAs, there is:

1) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints,
2) a small effect on the available SDM, and
3) a small effect on the ejected CEA worth used in the accident analysis.

Therefore, a one hour time period is sufficient to:

1) identify causes of a misaligned CEA,
2) take appropriate corrective action to realign the CEAs, and
3) minimize the effects of xenon redistribution.

With a large CEA misalignment (≥ [19 inches]) however, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on:

1) the available SDM,
2) the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, and
3) the ejected CEA worth used in the accident analysis.

The one-hour Completion Time allowed is sufficient to identify causes of a misaligned CEA, take appropriate corrective action to realign the CEAs, and minimize the effects of xenon redistribution.

In Condition B, power operations may continue provided adequate SHUTDOWN MARGIN exists and the CEAs can be properly aligned. This may be accomplished by aligning the OPERABLE CEAs to the inoperable CEA (sub-group). Maintaining the insertion and sequence limits of LCO 3.1.7 ensures adequate SHUTDOWN MARGIN and proper power distribution are maintained. The two-hour Completion Time to align the OPERABLE CEAs provides the operator sufficient time to properly align the OPERABLE CEAs to the inoperable CEA (sub-group).

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BASES

ACTIONS (continued)

Although a part-strength CEA has less of an effect on core flux than a full -strength CEA, a misaligned part-strength CEA will still result in xenon redistribution and effect core power distribution. Requiring realignment within 1 hour minimizes these effects and ensures an acceptable power distribution is maintained.

C.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions without challenging plant systems.

D.1 and D.2

In the case of:

1) more than one inoperable CEA [sub-group] or,
2) more than one CEA misaligned from any other CEA in their group by more than [19] inches or,
3) one or more CEAs inoperable as the result of excessive friction or mechanical interferences or known to be untrippable, continued operation is not allowed. This is because either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, a loss of SDM.

If a CEA is inoperable as a result of excessive friction or mechanical interference or is untrippable, it is not available for reactivity insertion during a reactor trip. With an untrippable CEA, meeting the insertion limits of LCO 3.1.7 does not ensure that adequate SDM exists. In this condition, an additional allowance must be made for the worth of the affected CEA when calculating the available SDM.

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BASES

ACTIONS (continued)

This is necessary since the OPERABLE CEAs must still meet the single failure criteria. If additional negative reactivity is required to provide the necessary SDM, it must be provided by increasing the RCS boron concentration. One hour allows sufficient time to perform the calculation and make any required boron adjustment to the RCS.

The six hour Completion Time to reach MODE 3 is reasonable, based on operating experience, to reach MODE 3 from full power conditions without challenging plant systems.

LCO 3.1.6 - Shutdown CEA Insertion Limits

A.1

Accident analysis assumes that the Shutdown CEAs are fully withdrawn any time the reactor is critical. This ensures that 1) the minimum SDM is maintained and, 2) the potential effects of a CEA ejection accident are limited to acceptable limits. CEAs are considered fully withdrawn at [145 inches wd] since this position places them outside the active region of the core. The required Completion Time of one hour to fully withdraw the Shutdown CEA allows the operator adequate time to adjust the CEA in an orderly manner and is consistent with the Required Completion Time for Action A.1 in LCO 3.1.5, CEA Alignment.

B.1

With the SDM reduced by the insertion of a Shutdown CEA, the operator can no longer rely on the Regulating CEAs being above the Transient Insertion Limit to ensure adequate SDM exists. In this case a reactivity balance is required to verify the SDM. If additional negative reactivity is required to provide the necessary SDM, it must be provided by increasing the RCS boron concentration. One hour allows sufficient time to perform the calculation and make any required boron adjustments to the RCS.

(continued)

BASES

ACTIONS (continued)

B.2

When the Required Action of A.1 cannot be completed within the required Completion Time, a controlled shutdown must be commenced. The six-hour Completion Time is reasonable, based on operating experience, to reach MODE 3 from full power conditions without challenging plant systems.

LCO 3.1.7 - Regulating CEA Insertion Limits

A.1 and A.2

If the CEAs are inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24 hour period and the Short Term Steady State Insertion Limits are exceeded, peaking factors can develop which are of immediate concern.

Additionally, since the CEAs can be in this condition without misalignment, penalty factors are not inserted in the CPCs to compensate for the developing peaking factors. Verifying the Short Term Steady State Insertion Limits are not exceeded ensures that the peaking factors which do develop are within those allowed for continued operation. Fifteen minutes provides adequate time for the operator to verify if the Short Term Steady State Insertion Limits are exceeded.

Experience has shown that rapid power increases in areas of the core in which the flux has been depressed may result in fuel damage as the linear heat rate in those areas rapidly increases. Restricting the rate of THERMAL POWER increases to ≤ 5% RTP per hour following CEA insertion beyond the Long Term Steady State Insertion Limits ensures the power transients experienced by the fuel will not result in fuel failure.

B.1

With the Regulating CEAs inserted between the Long Term Steady State Insertion Limit and the Transient Insertion Limit and approaching the five EFPD or 14 EFPD limits the core is approaching the acceptable limits placed on operation with flux patterns outside those assumed in the long term burnup assumptions. In this case the CEAs must be returned to within the

(continued)

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BASES

ACTIONS (continued)

B.1

Long Term Steady State Insertion Limits or the core must be placed in a condition in which the abnormal fuel burnup can not continue. Two hours is a reasonable time to return the CEAs to within the Long Term Steady State Insertion Limits.

C.1 and C.2

With COLSS out-of-service, operation beyond the Short Term Steady State Insertion Limits can result in peaking factors that could approach the DNB or LPD trip setpoints. Eliminating this condition within two hours limits the magnitude of the peaking factors to acceptable levels. Restoring the CEAs to within the limit or reducing THERMAL POWER to an allowable level ensures acceptable peaking factors are maintained.

D.1

During power operations SHUTDOWN MARGIN requirements are met by maintaining the shutdown CEAs fully withdrawn and maintaining the regulating CEAs above the Transient Insertion Limit (also called the Power Dependent Insertion Limit).

Satisfying these conditions ensures that initial operating conditions are no more severe than the initial conditions assumed in the accident analysis and that the minimum required SHUTDOWN MARGIN exists. This includes an allowance for the most reactive CEA which is assumed to be withdrawn from the core. With the regulating CEAs inserted beyond the Transient Insertion Limit adequate SHUTDOWN MARGIN may not exist. In this case a reactivity calculation must be performed to determine the SHUTDOWN MARGIN. If additional negative reactivity is required to provide the necessary SHUTDOWN MARGIN, it must be provided by increasing the RCS boron concentration. One hour allows sufficient time to perform the calculation and make any required boron adjustments to the RCS.

(continued)

BASES

ACTIONS (continued)

D.2.1 and D.2.2

Operation beyond the Transient Insertion Limits results in a loss of SHUTDOWN MARGIN and excessive peaking factors. While boron addition to the RCS can ensure adequate SHUTDOWN MARGIN, the CEAs must be returned to above the Transient Insertion Limits to eliminate the peaking problem. This can be accomplished by either restoring the CEAs to within the insertion limits (Required Action D.2.1) or reducing THERMAL POWER to less than or equal to that allowed for the actual CEA insertion (Required Action D.2.2). Two hours provides a reasonable time to accomplish this while limiting the peaking factors to acceptable levels.

E.1

Performing SR 3.1.7.1, CEA Position, within one hour and every four hours thereafter ensures improper CEA alignments are identified before unacceptable flux distributions occur.

F.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown must be commenced. The allowed time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions without challenging plant systems. In MODE 3 the reactor is not critical and excessive power peaking cannot occur.

SURVEILLANCE REQUIREMENTS

LCO 3.1.5 - CEA Operability and Alignment

SR 3.1.5.1

Verification that individual CEA positions are within [seven] inches of all other CEAs in the group at a 12 hour frequency allows the operator to detect a CEA beginning to deviate from its expected position.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.5.2

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "full in" and "full out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. The 12-hour frequency is adequate when the CEA Motion Inhibit and Deviation Circuits are OPERABLE. This frequency has been shown to be acceptable through operating experience.

SR 3.1.5.3 and 3.1.5.4

Exercising individual CEAs every 92 days verifies that all CEAs continue to be OPERABLE even if they are not regularly moved. The 92-day frequency is consistent with the recommendations of NUREG-1366 (Ref. 5).

SR 3.1.5.5

Performance of a CHANNEL FUNCTIONAL TEST of each Reed Switch Position Transmitter channel ensures the channel is OPERABLE and capable of indicating CEA position over the entire strength of the CEA travel. Since this test must be performed when the reactor is shutdown, a refueling frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.1.5.6

Verification of CEA drop times determines that the maximum CEA drop time permitted is consistent with the assumed drop time used in the safety analysis. Measuring drop times prior to reactor criticality after reactor vessel head removal assures the reactor internals and Control Rod Drive Mechanism will not interfere with CEA motion or drop time. Performing the tests at a refueling frequency assures no degradation in these systems has occurred that would adversely affect CEA motion or drop time.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

LCO 3.1.6 - Shutdown CEA Insertion Limits

SR 3.1.6.1

Verification that each Shutdown CEA is fully withdrawn assumes they are available to provide reactor shutdown after criticality. Performing the surveillance prior to withdrawing the Regulating CEAs during an approach to criticality assumes the Shutdown CEAs are withdrawn shortly before they may be required for shutdown and also allows the operator time to halt the approach to criticality should a Shutdown CEA not be fully withdrawn.

Verification that individual Shutdown CEA positions are fully withdrawn at a 12 hour frequency allows the operator to detect a CEA beginning to deviate from its expected position.

LCO 3.1.7 - Regulating CEA Insertion Limits

SR 3.1.7.1

With the PDIL alarm OPERABLE, verification of each Regulating CEA group position every 12 hours is sufficient to detect CEA positions that may approach the limits, and to provide the operator with time to undertake the required actions should the sequence or insertion limits be found to have exceeded.

SR 3.1.7.2

Verification of the accumulated time of insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits assures the cumulative time limits are not exceeded. The 24 hour frequency ensures the operator identifies a time limit that is being approached before it is reached.

SR 3.1.7.3

Demonstrating the PDIL Alarm OPERABLE verifies that the PDIL Alarm is functional. The 31-day frequency has been shown to be acceptable through operating experience.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

LCO 3.1.8 - Part Strength CEA

Insertion Limits

SR 3.1.8.1

Verification of each part strength CEA group position every 12 hours is sufficient to detect PSCEA positions that may approach the insertion limits, and provide the operator sufficient time to undertake the required actions should the limits be exceeded.

REFERENCES

1. 10 CFR 50 Part 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors.

2. 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, Criterion 10, Reactor Design, Criterion 26, Reactivity Limits.

3. System 80+ CESSAR-DC, Chapter 15, Accident Analysis.

4. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application)."

5. Draft NUREG-1366, "Improvements to Technical Specifications."

16A.4.6 B 3.1.8 SPECIAL TEST EXCEPTIONS

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Special Test Exceptions

BASES

BACKGROUND

This Bases addresses the Limiting Conditions for Operation (LCOs) and Surveillance Requirements for the following LCOs:

3.1.8 Special Test Exception - Shutdown Margin
3.1.9 Special Test Exception - Moderator Temperature Coefficient, Group Height, CEA Insertion Limits, Power Distribution Limits and Center CEA Misalignment

These LCOs are required to limit the core power distribution and reactivity during PHYSICS TESTS. Since the limits protected by these LCOs share common requirements, their BASES are combined.

Section XI (Test Control) of 10 CFR 50, Appendix B (Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants), (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not violated during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant, as specified in 10 CFR 50, Appendix A, General Design Criterion 1 (Quality Standards and Records), (Ref. 2). Requirements for notification of the NRC for the purpose of conducting tests and experiments are specified in 10 CFR 50.59 (Changes, Tests, and Experiments), (Ref. 3).

Key objectives of a test program are to provide assurance that the facility has been adequately designed, to validate the analytical models used in design and analysis, to verify assumptions used for predicting plant response, to provide assurance that installation of equipment in the facility has been accomplished in accordance with design, and to verify that operating and emergency procedures are adequate (Ref. 5). Testing prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation is required to accomplish these objectives.

(continued)

BASES

BACKGROUND (continued)

The requirements for PHYSICS TESTS for initial and reload fuel cycles assure the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 6).

Physics Tests procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of testing required to ensure design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long-term power operation.

It is acceptable to suspend certain LCOs for Physics Tests because fuel damage criteria are not exceeded. Even if an accident occurs during Physics Test with one or more LCOs suspended, fuel damage criteria are preserved because the limits on power distribution and shutdown capability are maintained during Physics Tests.

APPLICABLE SAFETY ANALYSIS

Physics Tests include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are Total Planar Radial Peaking Factor, Total Integrated Radial Peaking Factor, Azimuthal Power Tilt and Axial Shape Index, which represent initial condition input (power peaking) to the accident analysis. Also involved are the movable control components (Shutdown and Regulating CEAs), which affect power peaking and are required for shutdown of the reactor.

The CESSAR-DC (Ref. 7), Chapter 14 defines requirements or initial testing of the facility, including Physics Tests. Requirements for reload fuel cycle Physics Tests are defined in ANSI/ANS-19.6.1-1985 (Ref. 6). Physics Tests for reload fuel cycles are given in Table 1 of ANSI/ANS-19.6.1-1985. Measurement of critical boron concentration, CEA group worths, reactivity coefficients, flux symmetry, and core power distribution are taken. Although these Physics Tests are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of Physics Tests possible or practical.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

This is acceptable as long as the fuel design criteria are not violated. When one or more of the limits specified in

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LCO 3.1.1	Shutdown Margin - ~~$T_{avg} \geq 210°F$~~
LCO 3.1.4	Moderator Temperature Coefficient
LCO 3.1.5	CEA Alignment
LCO 3.1.6	Shutdown CEA Insertion Limits
LCO 3.1.7	Regulating CEA Insertion Limits

are suspended for Physics Tests, the fuel design criteria are preserved as long as Linear Heat Rate remains within its limit. Therefore, LCO 3.1.9 places limits on allowable THERMAL POWER during Physics Tests and require Linear Heat Rate and DNB parameter be maintained within limits.

The individual LCOs governing CEA group height, insertion, and alignment, Axial Shape Index, Total Planar Radial Peaking Factor, Total Integrated Radial Peaking Factor and Azimuthal Power Tilt preserve the linear heat rate limits. Additionally, the LCOs governing RCS flow, T_c, and pressurizer pressure contribute to maintaining departure from nucleate boiling parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the linear heat rate and DNB parameter limits. The criteria for Loss of Coolant Accident (LOCA) are specified in 10 CFR 50 (Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors), (Ref. 1). The criteria for the loss of forced reactor coolant flow accident are specified in the CESSAR-DC (Section 15.3.2.1), (Ref. 8). Operation within the linear heat rate limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria.

During PHYSICS TESTS one or more of the LCOs that normally preserve the local heat rate and DNB parameters limits may be suspended. However, the results of the accident analysis are not adversely impacted if verification of linear heat rate and DNB parameters within their limits is verified while the LCOs are suspended. Therefore, surveillance requirements are placed as necessary to ensure linear heat rate and DNB parameters remain within limits during PHYSICS TESTS. Performance of these surveillance allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

PHYSICS TESTS meet the criterion for inclusion in technical specifications since the components and process variable LCOs suspended during PHYSICS TESTS meet criterion 1, 2 and 3 of the Interim Policy Statement as described in Reference 8.

LCO

LCO 3.1.8 provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

LCO 3.1.9 permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to:

1) measure CEA worth,
2) determine the reactor stability index and damping factor under xenon oscillation conditions,
3) determine power distributions for non-normal CEA configurations,
4) measure rod shadowing factors, and
5) measure moderator temperature and power coefficients.

Additionally, it permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient.

(continued)

BASES

APPLICABILITY

LCO 3.1.8 is applicable in MODES 2 and 3. Although CEA worth testing is conducted in MODE 2, during the performance of these tests sufficient negative reactivity is inserted to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the special test exemption allows limited operation in MODE 3 without having to borate to meet the SDM requirements of LCO 3.1.1.

LCO 3.1.9 is applicable in MODES 1 and 2 because the reactor must be critical at various THERMAL POWER levels to perform the PHYSICS TESTS described in the LCO section. Limiting the test power plateau to less than 85% of Rated Thermal Power (RTP) ensures linear heat rates are maintained within acceptable limits.

ACTIONS

LCO 3.1.8 - Special Test Exception - Shutdown Margin

A.1

With any full strength CEA not fully inserted and less than the minimum required reactivity equivalent available for trip insertion, or with all CEAs inserted and the reactor subcritical by the reactivity equivalent of the highest worth CEA, restoration of the minimum SDM requirements must be accomplished by increasing the RCS boron concentration. The required Completion Time to initiate boration allows the operator sufficient time to align the valves and start the boric acid pumps and is consistent with the Completion Time of LCO 3.1.1.

(continued)

ACTIONS (continued)

LCO 3.1.9 - Special Test Exception - Moderator Temperature Coefficient, Group Height, CEA Insertion Limits, Power Distribution Limits, and Center CEA Misalignment

A.1 and B.1

If THERMAL POWER exceeds the test power plateau or the linear heat rate requirements of LCO 3.2.1 are exceeded, THERMAL POWER must be reduced to restore the additional thermal margin provided by the reduced THERMAL POWER. The 15 minute completion time ensures prompt action is taken to reduce THERMAL POWER to within acceptable limits.

C.1

If Required Action A.1 or B.1 cannot be completed within the required Completion Time, the reactor must be placed in MODE 3. This increases thermal margin and is consistent with the Required Actions of the Power Distribution LCOs. The required Completion Time of six hours is adequate to perform a controlled shutdown and is consistent with the Power Distribution LCO Completion Times.

SURVEILLANCE REQUIREMENTS

LCO 3.1.8 - Special Test Exception - Shutdown Margin

SR 3.1.8.1

Verification of the position of each partially or fully withdrawn full-strength is necessary to ensure the minimum negative reactivity requirements for insertion on a trip are preserved. A two-hour frequency has been proven to be acceptable through operating experience.

SR 3.1.8.2

Prior demonstration that each CEA to be withdrawn from the core during PHYSICS TESTS is capable of full insertion when tripped from at least 50% withdrawn provides assurance that the CEA will insert on a trip signal. The seven day requirement ensure the CEAs are operable and has been proven to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

LCO 3.1.9 - Special Test Exceptions - Moderator Temperature Coefficient, Group Height, CEA Insertion Limits, Power Distribution Limits, and Center CEA Misalignment

SR 3.1.9.1

Monitoring linear heat rate and DNBR ensures the linear heat rate and DNBR limits are not exceeded. Refer to B 3.2.1, Linear Heat Rate and B 3.2.4, DNBR, for a discussion of the Bases for these parameters. Continuous monitoring is accomplished by the COLSS which generates margin limits based on linear heat rate and DNBR and will generate a COLSS margin alarm should a limit be exceeded. This surveillance is not applicable at < 20% RTP because adequate linear heat rate and DNBR margins exist up to this power level.

SR 3.1.9.2

Verifying THERMAL POWER is equal to or less than that allowed by a test power plateau as specified in the PHYSICS TEST procedure, ensures adequate linear heat rate and DNBR margins are maintained while LCOs are suspended. The 1 hour frequency is sufficient based upon slow rate of power change and increased operational controls in place during PHYSICS TESTS. Monitoring linear heat rate ensures the limits are not exceeded. Refer to B 3.2.1, Linear Heat Rate and B 3.2.4, DNBR, for a discussion of the Bases for these parameters. This surveillance is not applicable at < 20% of RTP because adequate linear heat rate margin exists at this power level, and because the incore monitoring system is not available below 20% RTP.

The SR is modified by a Note which states that the SR is only applicable in MODE 1 ≥ 20% RTP. This is because the Fixed Incore Detector Monitoring System is not available below 20% RTP.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.10.2

Verifying THERMAL POWER is equal to or less than that allowed by a test power plateau as specified in the PHYSICS TEST procedure, ensures adequate linear heat rate and DNB parameter margins are maintained while LCOs are suspended. The one hour frequency is sufficient based upon slow rate of power change and increased operational controls in place during PHYSICS TESTS.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI (Test Control), Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants.

2. 10 CFR 50, Appendix A, General Design Criterion 1, Quality Standards and Records.

3. 10 CFR 50.59, Changes, Tests, and Experiments.

4. 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors.

5. Regulatory Guide 1.68, Revision 2, Initial Test Programs for Water-Cooled Nuclear Power Plants, U.S. Nuclear Regulatory Commission, August, 1978.

6. ANSI/ANS-19.6.1-1985, Reload Startup Physics Tests for Pressurized Water Reactors, American National Standards Institute, December 13, 1985.

7. System 80+ CESSAR-DC, Chapter 14, Testing Requirements and Chapter 15, Accident Analyses.

8. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application)."

B 3.1.10 Special Test Exception-CEDMS Testing

Background

CEDMS testing is performed before startup to verify the operability of the control element drives. Since this test requires the withdrawal of CEAs, the shutdown margin is reduced. In order that the test may be performed, this special test exception is provided since the requirements of LCO 3.1.1 would be too restrictive to allow performance of the test.

Applicable
Safety Analysis

In Ref. 1, the conditions of the CEDMS testing were analyzed. It was found that sufficient subcriticality is maintained in case of a CEA ejection accident. This is from the fact that prior to testing K(n-1) must be less than 0.99. The margin will preclude inadvertent criticality.

LCO

Suspension of the shutdown margin requirement of LCO 3.1.1 may be suspended for pre-startup testing of the CEDMS if four conditions are met. First, only one CEA may be withdrawn at a time. Second, no CEA may be withdrawn more than seven inches. Third, with RTCBs open, K(n-1) must be less than 0.99 before the start of testing. Fourth, all other operations which involve a reactivity increase must be suspended during testing.

Applicability

LCO 3.1.10 is applicable during MODES 4 and 5 since these are the modes during which CEDMS testing is performed.

Actions

A.1

If any of the four requirements are not met then testing must be suspended and the shutdown margin must be restored to the limit of LCO 3.1.1. This action is necessary for the prevention of an inadvertent criticality.

Surveillance
Requirements

SR 3.1.10.1

Determination of the shutdown margin ensures that CEDMS testing is being performed under conditons that would prevent an inadvertent criticality. The frequency of 24 hours is based upon operating experience and the fact that other administrative controls exist to prevent unauthorized reactivity increases.

B 3.1.10 (continued)

Reference

1. Safety Evaluation by the Office of NRR, Docket no. STN 50-530, January 26, 1988.
2. CESSAR-DC, Section 15.B "Shutdown Risk Report"

B 3.1.11 Boron Dilution Alarms

Background

There are two startup channel high neutron flux alarms in the System 80+ design. These alarms exist for the purpose of alerting the operator to an inadvertent boron dilution and subsequently the prevention of an inadvertent criticality.

Applicable Safety Analysis

In Ref. 1 it is mentioned that the use of two high neutron flux alarms provide proper redundancy for the detection of a boron dilution event. A single alarm failure will still leave the operator will adequate high neutron flux detection capability.

LCO

The LCO requires that both startup channel high neutron flux alarms shall be operable.

Applicability

LCO 3.1.11 is applicable during MODES 3, 4, 5, and 6. Since the reactor is critical in MODE 1 and also is critical (or approaching critical) in MODE 2, this LCO does not apply in MODES 1 and 2.

Actions

A.1 and A.2

With one startup channel high neutron flux alarm inoperable, action must be immediately taken to restore the inoperable channel to operable status. Also the RCS boron concentration must be determined when entering MODE 3, 4, 5, or 6 or at the time the alarm is determined inoperable. This second action is to be performed immediately and once per the frequency given in the LCO tables 3.1.11-1 through 3.1.11-5. These actions ensure that an alternate means is available for the detection of an inadvertent boron dilution event.

B.1, B.2, and B.3

With both startup channel high neutron flux alarms inoperable, action must be immediately taken to restore a single channel to operable status. Also the RCS boron concentration must be determined when entering MODE 3, 4, 5, or 6 or at the time the alarm is determined inoperable. This second action is to be performed immediately and once per the frequency given in the LCO tables 3.1.11-1 through 3.1.11-5. Immediate suspension of all operations involving core alterations or positive reactivity changes is also required. These actions will help prevent the loss of shutdown margin and return to criticality should an inadvertent

boron dilution event occur.

Surveillance
Requirements

SR 3.1.11.1

A channel check shall be performed on each startup channel once per 12 hours to ensure proper operation. The frequency is based upon operating experience and other administrative controls.

SR 3.1.11.2

A channel calibration shall be performed on each startup channel every 31 days of cumulative operation during shutdown. The frequency is based upon operating experience.

Reference

1. PVNGS Technical Specification Bases, 3/4.1.2.7.

2. CESSAR-DC, Section 19.8 "Shutdown Risk Report"

16A.5 B 3.2 POWER DISTRIBUTION LIMITS

B 3.2 POWER DISTRIBUTION LIMITS

BASES

BACKGROUND

This Bases addresses the limiting Conditions for Operation (LCOs) and Surveillance Requirements for the following LCOs:

3.2.1 Linear Heat Rate
3.2.2 Planar Radial Peaking Factor (F_{xy})
3.2.3 Azimuthal Power Tilt (T_q)
3.2.4 Departure from Nucleate Boiling Ratio (DNBR)
3.2.5 Axial Shape Index (ASI)

These LCOs are required to limit the core power distribution to the initial values assumed in the accident analysis. Since the limits protected by these LCOs share common requirements, their Bases are combined.

Operation within the limits imposed by these LCOs limits potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss of Coolant Accident (LOCA), loss of flow, ejected Control Element Assembly (CEA) or other accident requiring termination by a Reactor Protection System (RPS) trip function. These LCOs limit the amount of damage to the fuel cladding during an accident by assuring the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the power distribution include:

1. the use of full-strength (or part-strength) CEAs to alter the axial power distribution;

2. decreasing CEA insertion by boration, thereby improving the radial power distribution; and

3. correcting off-optimum conditions which cause margin degradations (e.g., CEA drop or misoperation).

(continued)

BASES

BACKGROUND (continued)

The core power distribution is controlled so that, in conjunction with other core operating parameters (CEA insertion and alignment limits), the power distribution does not result in violation of the LCOs. LCOs and Limiting Safety System Setpoints (LSSS) are based on the accident analysis (Ref. 3), so that Specified Acceptable Fuel Design Limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on linear heat rate and Departure from Nucleate Boiling (DNB).

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 1, 2). This is accomplished by maintaining the power distribution and coolant conditions so that the peak linear heat rate and DNB parameters are within operating limits supported by the accident analysis (Ref. 3) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F, (Ref. 1). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy-water reaction.

DNB occurs when the local heat flux addition (BTU per unit cladding surface area per unit time) from the fuel rods to the reactor coolant causes nucleate boiling to be replaced by a steam film along regions of the cladding. This condition results in a decrease in the heat transfer rate from the cladding to the coolant. This results in a large difference between the cladding surface temperature and the coolant saturation temperature. Inside the steam film, high cladding temperatures are reached, and a Zircaloy-water reaction may take place. The chemical reaction may cause failure by oxidation of the cladding, allowing an uncontrolled release of the radioactive fission products to the reactor coolant.

(continued)

BASES

BACKGROUND (continued)

Proximity to the DNB condition is expressed by the DNBR, defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and anticipated operational occurrences (AOOs) is limited to the value given by the CE-1 Correlation (Ref. 4) and is accepted as an appropriate margin to DNB for operating conditions.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) or the Core Protection Calculators (CPCs), monitor the core power distribution and are capable of verifying that the linear heat rate and the

DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak linear heat rate and DNBR. A DNBR penalty factor is included in the COLSS and CPC DNBR calculation to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burn up assemblies will experience less rod bow. In design calculations, the penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

The COLSS will indicate continuously to the operator how far the core is from the operating limits and provide an audible alarm should an operating limit be exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded (for AOOs) by initiating a reactor trip.

(continued)

BASES

BACKGROUND (continued)

COLSS continually generates an assessment of the margin to linear heat rate and DNBR operating limits. The data required for these assessments include measured in-core neutron flux data, CEA positions, and coolant inlet temperature, pressure, and flow.

In addition to the monitoring performed by COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margin by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies to indicate CEA position. In this case, the CPCs assume a minimum core power of [20%] rated thermal power (RTP). The [20% RTP] threshold is necessary because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high local power density or low DNBR trips in the RPS will initiate a reactor trip prior to exceeding fuel design limits.

The limits on ASI, F_{xy} and T_q, represent limits within which the linear heat rate and DNBR algorithms are valid. These limits are obtained directly from the core analysis.

APPLICABLE SAFETY ANALYSIS

The fuel cladding must not sustain damage as a result of operation (Condition 1) and AOOs (Condition 2), (Ref. 3). The Power Distribution and CEA Insertion and Alignment LCOs preclude core power distributions from occurring that would violate the following fuel design criteria:

1. During a large break LOCA, peak cladding temperature must not exceed a limit of 2200°F, 10 CFR 50.46, (Ref. 1).

2. During a loss of flow accident, there must be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a DNB condition; General Design Criterion 10 (Ref. 2). This is referred to hereafter as the 95/95 criterion.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Linear Heat Rate, Axial Shape Index and RCS Condition LCOs ensure these criteria are met as long as the core is operated within the ASI, F^T_{xy}, F^T_r, and T_q limits specified in the LCO. The latter are process variables that characterize the three-dimensional power distribution of the reactor core. Operation within the limits for these variables ensures their actual value is within the range used in the accident analysis.

Fuel cladding damage does not occur from conditions outside these LCOs during normal operation. However, fuel cladding damage could result should an accident occur from initial conditions outside the limits of these LCOs. Changes in the power distribution can cause increased power peaking and correspondingly increased local linear heat rates.

ASI, Linear Heat Rate, Planar Radial Peaking Factors, Azimuthal Power Tilt and Axial Shape Index satisfy criterion 2 of the Interim Policy Statement as described in Reference 5.

LCO

The power distribution LCO limits are based on correlations between power peaking and the measured variables used as inputs to the linear heat rate and DNBR operating limits.

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

The limitation on DNBR as a function of ASI represents a conservative envelope of operating conditions consistent with the analysis assumptions which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR for all AOOs. Of these, the loss of flow transient is the most limiting. Operation of the core with a DNBR at, or above, this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Limiting the values of the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPCs to values equal to or greater than the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) provides assurance that the limits calculated by the COLSS and CPCs remain valid.

(continued)

BASES

LCO (continued)

The limitations on T_q are provided to ensure that design operating margins are maintained. A T_q greater than 0.10 is not expected. Should it occur, the Actions ensure operation is restricted to only those conditions required to identify the cause of the tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untitled power distribution. lt

The limitation on ASI ensures that the actual ASI value is maintained within the range of values used in the accident analysis. The ASI limits ensure that with Tq at its maximum upper limit, DNBR will not drop below the DNBR Safety Limit for AOOs.

Exception to Fxy, Tq and DNBR requirements are allowed during physics testing provided the limits described in LCO 3.1.9 are met

ACTIONS

LCO 3.2.1 - Linear Heat Rate

A.1 and A.2

Operation at or below the COLSS calculated power limit based on kw/ft assures that the linear heat rate limit is not exceeded. If the COLSS calculated core power limit based on kw/ft exceeds the operating limit, initiating corrective action within 15 minutes ensures prompt action is taken to reduce it below the limit. One hour is a reasonable time to accomplish this.

B.1 and B.2

If COLSS is not available, the OPERABLE local power density channels are monitored to ensure the linear heat rate limit is not exceeded. Operation within this limit ensures that in the event of a LOCA, the peak temperature of the fuel clad will not exceed 2200°F. If the linear heat rate limit is exceeded, initiating corrective action within 15 minutes ensures prompt action is taken to reduce it below the limit. Two hours is a reasonable time to restore it to within limits when COLSS is not in use. If, during this time, conditions develop that would approach core safety limits, a reactor trip will be generated by the CPCs.

(continued)

BASES

ACTIONS (continued)

C.1

If linear heat rate cannot be returned to within its limit, core power must be reduced. Reduction to less than 20% RTP ensures the core is operating further from thermal limits and places the core in a conservative condition based on the trip setpoints generated by the CPCs which assume a minimum core power of 20% RTP. Six hours is a reasonable time, based on operating experience, to reach 20% RTP without challenging plant systems.

LCO 3.2.2 - Planar Radial Peaking Factors (F_{xy})

A.1.1, A.1.2 and A.2

If the measured F^{m}_{xy} exceed the values in the COLSS and CPCs, non-conservative operating limits and trip setpoints may be calculated. In this case, action must be taken to ensure the COLSS operating limits and ~~PC~~ CPC trip setpoints remain valid with respect to the accident analysis. The operator can do this by accomplishing the Required Actions. Six hours provides adequate time to change these values while limiting the time the plant is operated in this condition.

A.3

If A.1.1 and a.1.2, or A.2 cannot be accomplished, core power must be reduced. Reduction to less than 20% RTP ensures the core is operating further from thermal limits and places the core in a conservative condition based on the trip setpoints generated by the CPCs which assume a minimum core power of 20% RTP. Six hours is a reasonable time, based on operating experience, to reach 20% RTP without challenging plant systems.

LCO 3.2.3 - Azimuthal Power Tilt - T_q

(continued)

BASES

ACTIONS (continued)

A.1 and A.2

If the measured T_q is greater than the T_q allowance used in the CPCs, non-conservative trip setpoints may be calculated. If the T_q is restored, the reactor may return to normal operation. Two hours is sufficient time to allow the operator to reposition CEAs and significant radial xenon redistribution will not occur within this time. If the T_q cannot be restored within two hours, the T_q allowance in the CPCs must be adjusted to equal to or greater than the measured value to ensure the design safety margins are maintained.

B.1, B.2.2.1, B.2.2.2 and B.2.2.3

With $T_q > 0.10$, T_q must be returned to within limits to ensure acceptable flux peaking factors are maintained. Operation may proceed for up to two hours while attempts are made to restore T_q to within its limit.

If T_q cannot be returned to within limits, it may still be desirable to retain the ability to operate the reactor. In the case of a tilt generated by a CEA misalignment it allows recovery of the CEA while continuing to operate. Except as a result of CEA misalignment, a T_q of greater than 0.10 is not expected. However, if it should occur, continued operation of the reactor may be necessary to discover the cause of the tilt. If this occurs, operation is restricted to only those conditions required to identify the cause of the tilt. It is necessary to explicitly account for power asymmetries because the radial power peaking factors used in the core power distribution calculations are based on an untilted power distribution.

If the T_q limits are not restored the reactor will continue to operate with an axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation, which will result in increased linear heat generation rates when the xenon redistributes. If T_q can not be restored to within its limit within two hours, reactor power must be reduced. Reducing THERMAL POWER to < [50% RTP] within four hours provides conservative protection from increased peaking due to potential xenon redistribution. Required Action B is modified by two Notes which requires

(continued)

BASES

ACTIONS (continued)

B.1, B.2.2.1, B.2.2.2 and B.2.2.3

all subsequent actions be performed once Required Action B is entered. This ensures corrective action is taken before unrestricted power operation resumes.

The linear power level - high trip setpoints are reduced to ≤[55% RTP] to ensure the assumption of the accident analysis are maintained.

THERMAL POWER is restricted to [50% of RTP] until T_q is restored to within its limit. This Action prevents the operator from increasing THERMAL POWER above the conservative limit when a significant T_q has existed, but allows the unit to continue operation for diagnostic purposes.

The Completion Time of Required Action B.2.2.3 is modified by a Note governing subsequent power increases. After a THERMAL POWER increase following restoration of T_q to within its limit, operation may proceed provided T_q is determined to remain within its limit at the increased THERMAL POWER level.

The provision to allow discontinuation of the surveillance after verifying T_q is acceptable allows exit from the Action after T_q has been returned to an acceptable value.

C.1

If T_q cannot be restored within its limit, core power must be reduced. Reduction to less than 20% RTP ensures the core is operating further from thermal limits and places the core in a conservative condition based on the trip setpoints generated by the CPCs which assume a minimum core power of 20% RTP. Six hours is a reasonable time, based on operating experience, to reach 20% RTP without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

LCO 3.2.4 - DNBR

A.1, B.1, A.2 and B.2

Operating the core at or above the minimum required value of DNBR provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient. If the core power exceeds the core power limit based on DNBR, fuel design limits may not be maintained following a loss of flow and prompt action must be taken to restore DNBR to above the minimum allowable value. Fifteen minutes is a reasonable time to allow the operator to initiate controlled actions to restore DNBR. With COLSS in service, the actions to restore DNBR should be completed within one hour to limit the time the plant is operated outside the initial conditions assumed in the analyses. With COLSS out of service, two hours is allowed because of added conservatisms in the CPCs.

C.1

If DNBR cannot be restored within the allowed times, core power must be reduced. Reduction to less than 20% RTP ensures the core is operating further from thermal limits and places the core in a conservative condition based on the trip setpoints generated by the CPCs which assume a minimum core power of 20% RTP. Six hours is a reasonable time, based on operating experience, to reach 20% RTP without challenging plant systems.

LCO 3.2.5 - Axial Shape Index

A.1

The ASI limits preserve the LOCA and loss of flow accident criteria assumed in the accident analysis.

Two limits for ASI are provided, one with COLSS OPERABLE, the other with COLSS out of service. The COLSS limit is less restrictive because of the greater accuracy of the incore neutron detector system. If ASI exceeds its limit, two hours is allowed to restore ASI to within the limit. This gives

(continued)

BASES

ACTIONS (continued)

A.1

the operator time to reposition the regulating or part-strength CEAs to reduce the axial power imbalance. The magnitude of any potential xenon oscillation is reduced if the condition is not allowed to persist for more than two hours.

B.1

If the ASI is not restored to within its limits, the reactor will continue to operate with an axial power distribution mismatch. Continued operation in this configuration induces an axial xenon oscillation, and results in increased linear heat generation rates when the xenon redistributes. Reducing THERMAL POWER to less than 20% of RTP reduces the maximum linear heat rate to a value that will most likely not exceed the fuel design limits should an event occur. Six hours is a reasonable time, based on operating experience, to reach 20% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

LCO 3.2.1 - Linear Heat Rate

SR 3.2.1.1

Continuous monitoring of linear heat rate is provided by the COLSS which calculates core power and core power operating limits based on linear heat rate and continuously displays these limits to the operator. A COLSS margin alarm is annunciated in the event THERMAL POWER exceeds the core power operating limit based on linear heat rate.

The SR is modified by a Note which states SR 3.0.4 does not apply because surveillance cannot be accomplished prior to increasing THERMAL POWER above 20% RTP because COLSS assumes a minimum THERMAL POWER of 20% RTP.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.2

With the COLSS out of service, the operator must monitor linear heat rate with the local power density channels. A two-hour frequency is adequate to allow the operator to identify trends that would result in conditions that would approach to the linear heat rate limits.

The SR is modified by a Note which states SR 3.0.4 does not apply because surveillance cannot be accomplished prior to increasing THERMAL POWER above 20% RTP because COLSS assumes a minimum THERMAL POWER of 20% RTP.

SR 3.2.1.3

Verification that the COLSS margin alarm actuates at a power level equal to or less than the core power operating limit based on kw/ft ensures the operator will be alerted should operating conditions approach the kw/ft operating limit. A 31-day frequency is typical of the frequency of functional tests, and has been shown to be acceptable through operating experience.

The SR is modified by a Note which states SR 3.0.4 does not apply because surveillance cannot be accomplished prior to increasing THERMAL POWER above 20% RTP because COLSS assumes a minimum THERMAL POWER of 20% RTP.

LCO 3.2.2 - Planar Radial Peaking Factors

SR 3.2.2.1

The periodic surveillance requirements for determining the calculated Planar Radial Peaking Factor provides assurance that the Planar Radial Peaking Factor used in the COLSS and CPCs remain valid throughout the fuel cycle. Determining the measured Planar Radial Peaking Factors after each fuel loading prior to exceeding 70% RTP provides additional assurance that the core was properly loaded.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

Performance of the surveillance every 31 EFPD ensures unacceptable changes in the Planar Radial Peaking Factor are promptly detected.

The SR is modified by a Note which states SR 3.0.4 does not apply because surveillance cannot be accomplished prior to increasing THERMAL POWER above 20% RTP because COLSS assumes a minimum THERMAL POWER of 20% RTP.

LCO 3.2.3 - Azimuthal Power Tilt

SR 3.2.3.1

Continuous monitoring of T_q by the incore neutron detectors is provided by the COLSS. A COLSS alarm is annunciated in the event the measured T_q exceeds the value used in the CPCs.

SR 3.2.3.1

With the COLSS out of service, the operator must calculate the T_q. A 12-hour frequency allows the operator to identify developing tilts, although tilt causing events should be detectable in their own right.

SR 3.2.3.3

Verification that the COLSS Azimuthal Tilt alarm actuates at a value less than the value used in the CPCs ensures the operator will be alerted should T_q approach its operating limit. A 31-day frequency is typical of the frequency of functional tests.

SR 3.2.3.4

Independent confirmation of the validity of the COLSS calculated T_q ensures the COLSS will accurately identify T_qs. A 31-day frequency is based on engineering judgment and has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

LCO 3.2.4 - DNBR

SR 3.2.4.1

Continuous monitoring of DNBR is provided by the COLSS which calculates core power and core power operating limits based on DNBR and continuously displays these limits to the operator. A COLSS margin alarm is annunciated in the event THERMAL POWER exceeds the core power operating limit based on DNBR.

The SR is modified by a Note which states SR 3.0.4 does not apply because surveillance cannot be accomplished prior to increasing THERMAL POWER above 20% RTP because COLSS assumes a minimum THERMAL POWER of 20% RTP.

SR 3.2.4.2

With the COLSS out of service, the operator must monitor DNBR with the OPERABLE DNBR channels of the CPCs. A two-hour frequency is adequate to allow the operator to identify trends in conditions that would result in an approach to the DNBR limit.

The SR is modified by a Note which states SR 3.0.4 does not apply because surveillance cannot be accomplished prior to increasing THERMAL POWER above 20% RTP because COLSS assumes a minimum THERMAL POWER of 20% RTP.

SR 3.2.4.3

Verification that the COLSS margin alarm actuates at a power level equal to or less than the core power operating limit based on DNBR ensures the operator will be alerted should operating conditions approach the DNBR operating limit. A 31-day frequency has been typical of the frequency of functional tests.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

The SR is modified by a Note which states SR 3.0.4 does not apply because surveillance cannot be accomplished prior to increasing THERMAL POWER above 20% RTP because COLSS assumes a minimum THERMAL POWER of 20% RTP.

LCO 3.2.5 - Axial Shape Index

SR 3.2.5.1

ASI can be monitored by both the incore (COLSS) and excore (CPC) neutron detector systems. COLSS provides the operator with an alarm should an ASI limit be approached.

Verification of ASI every 12 hours ensures the operator is aware of changes in ASI as they develop. A 12-hour frequency for surveillance is acceptable because the mechanisms which affect ASI, such as xenon redistribution or CEA drive mechanism malfunctions, cause slow ASI changes and will be discovered before the limits are exceeded.

REFERENCES

1. 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors.
2. 10 CFR 50.46, Appendix A, General Design Criteria for Nuclear Power Plants.
3. System 80+ CESSAR-DC, Chapter 15, Accident Analysis and Chapter 6, Engineered Safety Features.
4. C-E1 Correlation for DNBR.
5. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Technical Specifications-Volume 1 (Criteria Application)."

16A.6 B 3.3 INSTRUMENTATION

16A.6.1 B 3.3.1 RPS INSTRUMENTATION: PRESSURIZER, CONTAINMENT, STEAM GENERATOR, REACTOR COOLANT FLOW, LOSS OF LOAD

B 3.3 INSTRUMENTATION

LCO 3.3.1 Reactor Protective System (RPS) Instrumentation:
Pressurizer, Containment, Steam Generator, Reactor Coolant Flow, Loss Of Load

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during Anticipated Operational Occurrences (AOOs), and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

Four measurement channels with electrical and physical separation are provided for each parameter used in the direct generation of trip signals, with the exception of the control element assembly (CEA) position indication used in the Core Protection Calculators (CPCs). When any two channels of like instrumentation receive a trip signal, a reactor trip is generated. The reactor trip circuit breakers open, power to the CEAs is interrupted, and the CEAs fall into the core.

A 2/3 trip logic is all that is required to provide adequate plant protection, since a failure of one channel still ensures a reactor trip would be generated by the two remaining OPERABLE channels. When performing maintenance, testing, or removing a failed channel from service, RPS logic for the affected parameter(s) is changed from 2/4 to 2/3 by trip channel bypassing the affected trips. All RPS trips can be trip channel bypassed, providing each is bypassed in one RPS channel at a time.

The trip signal is generated by the Bistable Logic processors which compare the input signals to either fixed or variable setpoints. These Bistable outputs for each parameter (e.g. Pressurizer Pressure, Steam Generator Level etc.) are sent to Local Coincidence Logic where the two-out-of four logics are performed.

The trip channel bypasses and operating bypasses are manipulated by separate Interface and Test processors.

(continued)

BASES

BACKGROUND (continued)

The trip channel bypass prevents a bistable trip from contributing to the initiation of protective action. The trip channel bypass information is provided to four channels of Local Coincidence Logics to change their logic into 2/3 by Interface and Test processors. The LCLs only allow one channel bypass at a time.

In addition to the trip channel bypasses, there are also operating bypasses on selected RPS trips. These bypasses are enabled manually, in all four RPS channels, when plant conditions do not warrant the specific trip protection. All operating bypasses are automatically removed when enabling bypass conditions are no longer satisfied.

Since a single failure neither causes nor prevents the protection system actuation, and protection channels are isolated from control and monitoring channels via fiber optics cabling, this arrangement meets the requirements of IEEE 279 (Reference 2.).

APPLICABLE SAFETY ANALYSIS

The required channels of RPS Instrumentation provide plant protection during AOOs and assist the Engineered Safety Features (ESF) systems in the mitigation of certain accidents.

Pressurizer Pressure - High

The "Pressurizer Pressure - High" trip, in conjunction with the pressurizer safety valves and the main steam safety valves, provides protection against overpressurization of the RCS during the following events:

1. Loss of Electrical Load (AOO)
2. Loss of Condenser Vacuum (AOO)
3. CEA Withdrawal From Low Power Conditions (limiting RCS pressure case) (AOO)
4. Chemical and Volume Control System Malfunction (CVCS) (AOO)
5. Main Feedwater System Pipe Break (Accident)

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of Loss of Coolant Accidents.

Containment Pressure - High

The Containment Pressure - High trip is provided to generate a reactor trip to prevent exceeding the containment design pressure of [49] psig during a design basis LOCA or Main Steam Line Break (MSLB) accident.

Steam Generator Pressure - Low

The Steam Generator Pressure - Low trip provides protection against an excessive rate of heat extraction from the steam generators and resulting rapid, uncontrolled cooldown of the RCS. This trip is needed to shutdown the reactor and assist the ESF System in the event of a Main Steam Line Break, or Main Feedwater Line Break Between the Steam Generator and the Check Valve accidents.

Steam Generator Level - Low

The Steam Generator Level - Low trip ensures that a reactor trip signal is generated for the following events to help prevent exceeding the design pressure of the Reactor Coolant System due to the loss of the heat sink:

1. Loss of Normal Feedwater Event (AOO).

2. Feedwater System Pipe Break (Accident)

Steam Generator Level - High

The Steam Generator Level - High trip is provided to protect the turbine from excessive moisture carryover in the case of a steam generator overfill event, such as Excess Main feedwater (AOO) and Steam Generator Tube rupture transients.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection against a Reactor Coolant Pump Sheared Shaft Event and certain steam line break events with a concurrent loss of AC power. The DNBR limit is expected to be exceeded during this event; however, the trip ensures the consequences are acceptable.

The above trips meet Criterion 3 for inclusion as a technical specification as they are part of a system on the primary success path for reactivity control.

LCO

The LCO on RPS instrumentation channels ensures that each of the following requirements is met:

1. A reactor trip will be initiated when necessary.

2. The required protection system instrumentation coincidence logic is maintained (minimum 2/3 logic).

3. Sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance.

The Allowable Value specified ensure that violation of the core and RCS Safety Limits does not occur during normal operation and AOOs, and assist the ESF Actuation System in mitigating the consequences of accidents.

Only the Allowable Values are specified for each RPS trip function in the LCO. Each allowable value is specified such that the analytical limit assumed in the safety analysis is conservative including all applicable setpoint uncertainties.

Pressurizer Pressure - High

This trip is set below the lift setting of the pressurizer code safety valves and its operation avoids the undesirable operation of these valves during normal plant operation. This setpoint ensures the reactor trip will take place, thereby limiting further heat input to the RCS (and consequent pressure rise). The Pressurizer code safety valves lift to prevent overpressurization of the RCS.

(continued)

BASES

LCO (continued)

Pressurizer Pressure - Low

This trip is set low enough to prevent a reactor trip during normal plant operation and pressurizer pressure transients. However, the setpoint is high enough that with a Loss of Coolant Accident, the reactor trip will occur soon enough to allow the ESF systems to perform as expected in the analyses and mitigate the consequences of the accident. The setpoint may be manually decreased to a minimum value as pressurizer pressure is reduced during controlled plant shutdowns, provided the margin between the pressurizer pressure and the setpoint is maintained at the specified value. This allows for controlled depressurization of the RCS while still maintaining an active trip setpoint until the time is reached when the trip is no longer needed to protect the plant. The setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached.

The Pressurizer Pressure - Low trip setpoint may be manually (decrease to floor value, to allow for a controlled cooldown and depressurization of the RCS without causing a reactor trip, or a Safety Injection Actuation. The margin between the actual pressurizer pressure and the trip setpoint must be maintained at the specified value to ensure a reactor trip will occur if required during RCS cooldown and depressurization.

Containment Pressure - High

This trip is set high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup), and not indicative of an off-normal condition. It is set low enough to initiate a reactor trip when an off-normal condition is indicated. This allows the ESF systems to perform as expected in the Accident Analyses by mitigating the consequences of the analyzed accidents.

(continued)

BASES

LCO (continued)

Steam Generator Pressure - Low

This setpoint is sufficiently below the full load operating value for steam pressure so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of excessive steam demand. Since excessive steam demand causes the RCS to cool down resulting in positive reactivity addition to the core, a reactor trip is required to offset that effect.

The setpoint may be manually decreased as steam generator pressure is reduced during controlled plant cooldown, provided the margin between steam generator pressure and the setpoint is maintained at the specified value. This allows for controlled depressurization of the secondary system while still maintaining an active reactor trip setpoint and Main Steam Isolation Signal (MSIS) setpoint until the time is reached when the setpoints are no longer needed to protect the plant. The setpoint increases automatically as steam generator pressure increases until the specified trip setpoint is reached.

Steam Generator Level - Low

This setpoint is programmed such that as reactor power decreases the level setpoint is decreased from the normal full power value down to a minimum preset low power value.

This trip ensures that a reactor trip signal is generated to help prevent exceeding the design pressure of the RCS due to the loss of heat sink.

Steam Generator Level - High

This setpoint is high enough to allow for normal plant operation and transients without causing a reactor trip. It is set low enough to ensure a reactor trip occurs before the water level reaches the steam dryers.

Tripping the reactor on high steam generator water level, (in conjunction with a turbine trip on a reactor trip) prevents moisture carryover to the turbine.

(continued)

BASES

LCO (continued)

Reactor Coolant Flow - Low

The trip is set low enough to allow for the slight variations in reactor coolant flow during normal plant operations. However, it ensures that a reactor trip signal is generated in the event of a Reactor Coolant Pump Sheared Shaft in order to yield acceptable consequences. Tripping the reactor ensures that the resultant power-to-flow ratio provides adequate core cooling under the expected pressure conditions for the event.

The Reactor Coolant Flow - Low trip setpoint may be adjusted ~~to bypass the trip~~ when reactor power reaches the specified value. This allows for the de-energization of up to one ~~or more~~ Reactor Coolant Pump~~s~~ (RCP~~s~~) per SG loop (e.g. for plant cooldown), while maintaining the ability to keep the shutdown CEA banks withdrawn from the core if desired. LCOs 3.4.3., 3.4.4, 3.4.5, 3.4.6, and 3.4.7 (RC Loops and Circulation) ensure adequate RCS flowrate is maintained.

INSERT E

APPLICABILITY

Except for the SG Pressure-Low and RC Flow-Low, these trips are applicable in MODES 1 and 2 only because the reactor can be critical in these modes. The trips are designed to take the reactor subcritical which assists (as described above) in mitigating the consequences of the particular accidents and AOOs listed.

In MODES 3, 4, and 5, the main concern is ~~for~~ a return to power. ~~event~~. The reactor is protected ~~during this event~~ by and the High Log Power trip, and therefore, the ~~above~~ other trips do not need to be OPERABLE.

for events which could result in a return to power by the SG Pa-Low Trip the RCF Low-Low trip,

ACTIONS

If a protection channel of a given process variable becomes inoperable, the goal shall be to return the inoperable channel to service as soon as practical, but no later than prior to returning to MODE 2 following entry into MODE 5. This approach to bypass/trip in four channel protection systems is consistent with the applicable criteria of IEEE Standard 279.

NOTE

The Unit Review Group (URG) shall determine the desirability of maintaining any channel in bypass pursuant to Administrative Controls.

(continued)

INSERT E

The analyses of increased heat removal and CEA withdrawal events would show unacceptably low values of DNBR if they were to be initiated with less than one RCP operating in each steam generator loop.

BASES

ACTIONS (continued)

A.1

RPS coincidence logic is normally 2/4. If the number of channels OPERABLE is one less than the Total Number of Channels (i.e., only 3 out of 4 available), startup or power operation is allowed to continue as long as the inoperable channel is placed in bypass or trip within 1 hour. The provision of 4 independent and redundant trip channels allows one channel to be bypassed (removed from service) during operations, placing the RPS in 2/3 coincidence logic. It is preferable to place an inoperable channel in bypass rather than trip, since no single additional failure can spuriously trip the reactor.

If the channel fails (or is placed) in the tripped condition, just one spurious signal from any of the other three channels will cause the reactor to trip. Although this is acceptable from an accident analysis standpoint, it is not good operating practice. The time allotted to bypass or trip the channel allows the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

A.2

Some of the RPS instrumentation channels feed other RPS instrumentation channels and Engineered Safety Features Actuation System (ESFAS) instrumentation channels. If one of the RPS channels fails, the associated RPS and ESFAS channels should also be bypassed within 1 hour, for the same reasons, if it is determined that they are inoperable.

A.3

An inoperable channel must be returned to OPERABLE status prior to entry into MODE 2 following the next entry into MODE 5. This is in keeping with the general philosophy of not operating the plant in an off-normal condition indefinitely. The time allowed should be adequate to repair the inoperable channel. If the channel cannot be returned to OPERABLE status by the end of the next entry into MODE 5, the problem must be taken care of before startup and power operation can commence.

(continued)

ACTIONS (continued)

B.1

With the number of channels OPERABLE one less than the Minimum Channels Operable requirement, one inoperable channel must be placed in bypass, and the other channel must be placed in trip within the required Completion Time. With one channel of protective instrumentation bypassed, the RPS is in 2/3 logic, but with another channel failed, the RPS may be operating with a 2/2 logic (this assumes the channel failed in the non-conservative direction away from the trip setpoint). This is outside the assumptions made in the analyses and must be corrected. To correct the problem, the second channel is placed in trip. This places the RPS in a 1/2 logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

B.2

Some of the RPS instrumentation channels feed other RPS instrumentation channels and Engineered Safety Features Actuation System (ESFAS) instrumentation channels. If a second RPS channel fails, the associated RPS and ESFAS channels should also be placed in trip or bypass within the same Completion Time as B.1, if it is determined that they are inoperable.

Operation in MODES 1 and 2 may continue until the next CHANNEL FUNCTIONAL TEST. Operation in MODES 1 and 2 cannot continue beyond that time.

C.1

If action A or B cannot be completed within the required completion time, the reactor must be brought to a MODE where the action statements do not apply. The six hours is adequate to shutdown the plant in a controlled manner, while keeping the risk of operating in this condition at an acceptable level.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1

Performing a CHANNEL CHECK once per 12 hours ensures that any channel which drifts beyond the normal expected instrument drift, as compared to other identical channels, is detected within a reasonable amount of time.

The Data Processing System (DPS) continuously performs a cross channel comparison and will institute an alarm to warn operators that a channel has drifted out-of-tolerance or is not working properly.

The reason the CHANNEL CHECK is performed instead of a more thorough test, is that it imposes much less of a burden on operations than any of the other instrument channel tests. If the other instrument tests are performed at their appropriate frequencies, this test is very reliable.

SR 3.3.1.2

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire channel will perform its intended function when needed. It is not necessary to perform this test more often because operating experience has shown that the instrumentation stays quite reliable within that time frame.

Major portions of the Reactor Protection System are monitored and/or tested by the automatic test network. Those portions of the system which are not amenable to automatic testing because they involve actuation of electromechanical devices, or involve devices which are not within the PPS cabinets, can be tested manually. The automatic test network is capable of performing tests during reactor operation. The automatic testing does not degrade the ability of the RPS to perform its intended function.

Detail description of the RPS automatic test features are described in CESSAR-DC Chapter 7.2.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.3

Performance of a CHANNEL CALIBRATION every refueling ensures that the channels are reading accurately and within specified tolerances. Operating experience has shown that this test interval is satisfactory.

SR 3.3.1.4

Performance of a CHANNEL CALIBRATION every refueling ensures that the operating bypasses are operating accurately and within the specified tolerances. Operating experience has shown this test interval to be satisfactory.

SR 3.3.1.5

Verifying the logic for the operating bypasses to be OPERABLE within the specified time interval, ensures that the bypasses and their permissive setpoints function as designed prior to entering a MODE where they are required. Operating experience has shown the specified interval to be satisfactory.

REFERENCES

1. CESSAR-DC, Chapters 7 "Instrumentation and Controls", and 15 "Accident Analysis".

2. IEEE Standard 279-1971, Criteria for Protection Systems for Nuclear Power Generating Stations", April 5, 1972.

3. CESSAR-DC, Section 19.8 "Shutdown Risk Report"

16A.6.2 B 3.3.2 DEPARTURE FROM NUCLEATE BOILING RATIO (DNBR) AND ~~[illegible]~~ DENSITY (LPD) REACTOR PROTECTION SYSTEM (RPS) Ti
LOCAL POWER

DNBR & LPD
B 3.3.2

B 3.3 INSTRUMENTATION

B 3.3.2 Departure From Nucleate Boiling Ratio (DNBR) and Local Power Density (LPD) Reactor Protection System (RPS) Trips

BASES

BACKGROUND

DNBR and LPD are not directly sensed parameters (such as pressurizer pressure) but are a calculated composite of several inputs. The calculations are performed redundantly for each of the four RPS channels by separate core protection calculators (CPCs). The CPC channel outputs for the DNBR Low and LPD High trips are provided to four Bistable processors in the form of contact on or off to provide the designed 2/4 logic.

Each CPC receives several inputs and performs a number of calculations. To calculate the ratio of limiting to actual hot channel conditions in the core, the DNBR calculation considers:

1. Delta T power from coolant temperature and flow
2. Average neutron flux power
3. Axial power distribution
4. Radial peaking factors from CEA position measurement
5. Pressurizer pressure
6. Core inlet temperature
7. Coolant mass flow rate from reactor coolant pump speed

Low DNBR trip occurs when the calculated value reaches the DNBR Trip Setpoint. This DNBR calculation accounts for time delays and inaccuracies to prevent actual DNBR in the limiting channel from violating the minimum DNBR during all anticipated operating occurrences (AOOs).

(continued)

BASES

BACKGROUND (continued)

To calculate the current value of compensated peak power density, the LPD calculation considers:

1. Delta T power from coolant temperature and flow
2. Average neutron flux power
3. Axial power distribution
4. Radial peaking factors from CEA position measurement

High LPD trip occurs when the calculated value reaches the LPD Trip Setpoint. This setpoint prevents peak power densities from reaching levels that could produce fuel centerline melting by maintaining LPD at or below the Allowable Value.

A 2/3 trip logic is all that is required to provide adequate plant protection, since a failure of one channel still ensures a reactor trip would be generated by the two remaining OPERABLE channels. When performing maintenance, testing, or removing a failed channel from service, RPS logic for the trips is changed from 2/4 to 2/3 by trip channel bypassing the affected trip. All RPS trips can be trip channel bypassed, providing each is bypassed in one RPS channel at a time.

In addition to the trip channel bypass, there is also the % RTP operating bypass for the DNBR and LPD trips. This bypass is enabled, manually in all four RPS channels, when plant conditions do not warrant the trip protection. The operating bypass is automatically removed when enabling bypass conditions are no longer satisfied.

Since a single failure neither causes nor prevents the protection system actuation, and protection channels are isolated from control channels through fiber optic cables, this arrangement meets the requirements of IEEE 279 (Reference 2.).

(continued)

BASES

APPLICABLE SAFETY ANALYSIS

The DNBR and LPD trips provide plant protection during certain anticipated operating occurrences (AOOs), and assists the Engineered Safety Features (ESF) systems in the mitigation of certain accidents.

DNBR - Low

The "DNBR - Low" trip provides protection against core damage due to the occurrence of locally saturated conditions in the limiting (hot) channel during the following events:

1. Decrease in Feedwater Temperature
2. Increase in Feedwater Flow
3. Increased Main Steam Flow (not due to steam line rupture) Without Turbine Trip
4. Increased Main Steam Flow (not due to steam line rupture) With a Concurrent Single Failure of an Active Component
5. Steam Line Break with/without Concurrent Loss of Offsite AC Power
6. Loss of Normal AC Power
7. Partial Loss of Forced Reactor Coolant Flow
8. Total Loss of Forced Reactor Coolant Flow
9. Single Reactor Coolant Pump Shaft Seizure
10. Uncontrolled CEA Withdrawal From Low Power
11. Uncontrolled CEA Withdrawal at Power
12. CEA Misoperation; Full-Strength Subgroup CEA Drop

(continued)

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December 21, 1990

BASES

APPLICABLE SAFETY ANALYSES (continued)

13. CEA Misoperations; Full length or Part Strength CEA drop without RPCB.

14. Steam Generator Tube Rupture

15. Inadvertent depressurization of RCS

16. Uncontrolled boron dilution

17. Asymmetric Steam Generator Trip (ASGT)

18. Out-of-Sequence insertion or withdrawal of CEA group.

LPD - High

The "LPD - High" trip provides protection against fuel centerline melting due to the occurrence of excessive local flux density peaks during the following events:

1. Decrease in Feedwater Temperature

2. Increase in Feedwater Flow

3. Increased Main Steam Flow (not due to steam line rupture) Without Turbine Trip

4. Uncontrolled CEA Withdrawal From Low Power

5. Uncontrolled CEA Withdrawal at Power

6. CEA Misoperation; Single Full length or Part Strength CEA Drop

7. CEA Misoperation; Full-strength and part-strength CEA subgroup drop.

8. CEA Misoperation; Out-of-sequence operation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

For the events listed above (except CEA Misoperation) DNBR - Low will trip the reactor first since DNB would occur before fuel centerline melting would occur.

LCO

The Allowable Value specified ensure that violation of the Safety Limits for the reactor core and RCS is prevented during normal operations and AOOs, and assist the Engineered Safety Features (ESF) Actuation System in the mitigation of certain accidents.

Only the Allowable Values are specified for each RPS trip function in the LCO. Each allowable value is specified such that the analytical limit assumed in the safety analysis is conservative including all applicable setpoint uncertainties.

APPLICABILITY

The low DNBR and high LPD trips provide automatic protection functions when the reactor is critical. Because the reactor can be critical in either MODES 1 or 2, the LCO ensures that an RPS trip will occur when required, to prevent exceeding the SAFDLs during the AOOs listed, and help mitigate the consequences of the accidents listed.

The DNBR and LPD trips may be bypassed below the specified THERMAL POWER level because the trips are not needed below this power level to protect the core. The bypass shall be automatically removed if the permissive conditions are not met.

INSERT F →

The DNBR and LPD trips may also be bypassed during special testing. This is necessary in order to achieve the appropriate THERMAL POWER level and obtain necessary data for plant operation. The special tests contain power restraints which make bypassing these trips for this purpose acceptable.

(continued)

Amendment I
December 21, 1990

INSERT F

The function which automatically removes the CPC trip bypass at the specified THERMAL POWER level must be OPERABLE in Modes 3, 4, and 5, also, if the reactor trip breakers are closed. This trip function would provide a trip at two orders of magnitude lower power than the high log power trip when less than four RCPs were operating, thus preventing unacceptably low values of DNBR due to the combination of low flow and high power for increased heat removal and CEA withdrawal events.

ACTIONS

If a DNBR or LPD protection channel becomes inoperable, the goal shall be to return the inoperable channel to service as soon as practical, but no later than prior to returning to MODE 2 following entry into MODE 5. This approach to bypass/trip in four channel protection systems is consistent with the applicable criteria of IEEE standard 279.

The Unit Review Group (URG) shall determine the desirability of maintaining any channel in bypass pursuant to Administrative Controls.

A.1

RPS coincidence logic is normally 2/4. If the number of channels OPERABLE is one less than the Total Number of Channels (i.e., only 3 out of 4 available), startup or power operation is allowed to continue as long as the inoperable channel is placed in bypass or trip within 1 hour. The provision of 4 independent and redundant trip channels allows one channel to be bypassed (removed from service) during operations, placing the RPS in 2/3 coincidence logic. It is preferable to place an inoperable channel in bypass rather than trip, since no single additional failure can either spuriously trip the reactor, or prevent it from tripping.

If the channel fails (or is placed) in the tripped condition, just one spurious signal from any of the other three channels will cause the reactor to trip. Although this is acceptable from an accident analysis standpoint, it is not good operating practice. The time allotted to bypass or trip the channel allows the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

A.2

An inoperable channel must be returned to OPERABLE status prior to entry into MODE 2 following the next entry into MODE 5. This is in keeping with the general philosophy of not operating the plant in an off-normal condition indefinitely. The time allowed should be adequate to repair the inoperable channel. If the channel cannot be returned to OPERABLE status by the end of the next entry into MODE 5, the problem must be taken care of before startup and power operation can commence.

(continued)

BASES

ACTIONS (continued)

B.1

With the number of channels OPERABLE one less than the Minimum Channels Operable requirement, one inoperable channel must be placed in bypass, and the other channel must be placed in trip within the required Completion Time. With one channel of protective instrumentation bypassed, the RPS is in 2/3 logic, but with another channel failed, the RPS may be operating with a 2/2 logic (this assumes the channel failed in the non-conservative direction away from the trip setpoint). This is outside the assumptions made in the analyses and must be corrected. To correct the problem, the second channel is placed in trip. This places the RPS in a 1/2 logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

C.1

If action A or B cannot be completed within the required completion time, the reactor must be brought to a MODE where the action statements do not apply. The six hours is adequate to shutdown the plant in a controlled manner, while keeping the risk of operating in this condition at an acceptable level.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

Performing a CHANNEL CHECK every 12 hours ensures that any channel which is not reading the same (with allowed tolerances) as the other identical channels, is detected within a reasonable amount of time.

The Data Processing System (DPS) continuously performs a cross channel comparison and will initiate an alarm to warn operators that a channel has drifted out-of-tolerance or is not working properly.

The reason the CHANNEL CHECK is performed instead of a more thorough test, is that it imposes much less of a burden on operations than any of the other instrument channel tests. If the other instrument tests are performed at their appropriate frequencies, this test is very reliable.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.2

The RCS flow rate indicated by each CPC is verified to be less than or equal to the actual RCS total flow rate every 12 hours with THERMAL POWER ≥ 70% RTP. This check (and if necessary, the adjustment of the CPC addressable flow constant coefficients) ensures that the DNBR calculation is conservatively adjusted with respect to actual flow indications [as determined by the Core Operating Limit Supervisory System (COLSS)]. The interval is acceptable based upon engineering judgment and operating experience.

SR 3.3.2.3

A daily calibration including a heat balance (calorimetric), is performed when THERMAL POWER ≥ 20% RTP. The Linear Power Level signals and the CPC addressable constant multipliers are adjusted to make the CPC Delta T Power and Nuclear Power calculations agree with the calorimetric calculation if the absolute difference is ≥ 2%. These checks (and if necessary, the adjustment of the Linear Power Level signals and the CPC addressable constant coefficients) are adequate to ensure that the accuracy of these CPC calculations (and the resultant DNBR and LPD values) is maintained within the analyzed error margins with respect to reference indications. The interval is acceptable based upon engineering judgment and operating experience.

The excore neutron detectors are excluded from the CHANNEL CALIBRATIONS because of their inaccessibility, and the difficulty of simulating a meaningful neutron flux signal. Also, operating experience has shown them to be very reliable.

SR 3.3.2.4

The RCS flow rate indicated by each CPC is verified to be less than or equal to the actual RCS total flow rate every 31 days with THERMAL POWER ≥ 70% RTP. This check (and if necessary, the adjustment of the CPC addressable flow constant coefficients) ensures that the DNBR setpoint is conservatively adjusted with respect to actual flow indications as determined by a manual calorimetric calculation (the most accurate means of determining RCS flowrate). The interval is acceptable based upon engineering judgment and operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.5

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire channel will perform its intended function when needed. It is not necessary to perform this test more often because operating experience has shown that the instrumentation quite reliable within this time frame.

Major portions of the Reactor Protection System are monitored and/or tested by the automatic test network. Those portions of the system which are not amenable to automatic testing because they involve actuation of electromechanical devices, or involve devices which are not within the PPS cabinets, can be tested manually. The automatic test network is capable of performing tests during reactor operation. The automatic testing does not degrade the ability of the RPS to perform its intended function.

Detail description of the RPS automatic test features are described in CESSAR-DC Chapter 7.2.

SR 3.3.2.6

A CHANNEL FUNCTIONAL TEST including the injection of simulated process signals as close to the detectors as possible is performed every 18 months to ensure that the entire RPS trip channel, including the signal paths between the detectors and the RPS instrumentation, are able to perform their intended function. The test requires the reactor to be shutdown. More frequent performance of this test is not necessary because operating experience has shown the equipment to be reliable within this time frame.

SR 3.3.2.7

Performance of a CHANNEL CALIBRATION every 18 months ensures that the operating bypasses are operating accurately and within the specified tolerances. Operating experience has shown this test interval to be satisfactory.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.8

Verifying the logic for the operating bypasses to be OPERABLE within the specified time interval, ensures that the bypasses and their permissive setpoints function as designed prior to entering a MODE where they are required. Operating experience has shown the specified interval to be satisfactory.

SR 3.3.2.9

The vertical stack of three ex-core detectors in each channel is far enough from the core that the individual detectors are exposed to flux from all heights in the core. The CPCs adjust for this flux overlap by using the predetermined shape annealing matrix elements.

After refueling, it is necessary to reestablish the shape annealing matrix elements for the ex-core detectors based on more accurate incore detector readings. This is necessary because refueling could possibly produce a significant change in the annealed shape of the flux fields that the ex-core detectors are reading.

The reactor must be operating in the power range to have the excore linear power detectors be reading accurately. Also, power should be at a fairly significant value with somewhat steady state plant conditions to perform an accurate shape annealing calculation. For the above reasons, the surveillance frequency allows for the operator to make the determination of at what power level to make the adjustments, up to the specified limit. Operating experience has shown this interval to be acceptable.

REFERENCES

1. CESSAR-DC, chapters 7 "Instrumentation and Controls", and 15 "Accident Analysis".

2. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Generating Stations", April 5, 1972.

3. CESSAR-DC, section 19.8 "Shutdown Risk Report"

16A.6.3 B 3.3.3 VARIABLE OVERPOWER

B 3.3 INSTRUMENTATION

B 3.3.3 Variable Overpower

BASES

BACKGROUND

Linear power level is measured on the ex-core detector safety channels, and processed through amplification and bistable processors to provide reactor power indication and a variable overpower trip for the reactor protection system (RPS). The signals are processed redundantly for each of the four RPS channels by separate variable overpower channels.

Variable Overpower trip provides core protection against rapid reactivity excursions with the appropriate conservatism assumed in the transient and accident analysis. During normal operation, the trip setpoint follows the linear power level and maintained above the linear power level by STEP. The increase rate of the trip setpoint is limited by the RATE. During power excursion event if the linear power level increases greater than RATE and Exceeds the variable trip setpoint or is greater than CEILING, then trip signal is generated.

A 2/3 trip logic is all that is required to provide adequate plant protection, since a failure of one channel still ensures a reactor trip would be generated by the two remaining OPERABLE channels. When performing maintenance, testing, or removing a failed channel from service, RPS logic for the Variable Overpower - High trip is changed from 2/4 to 2/3 by trip channel bypassing. All RPS trips can be trip channel bypassed, providing each is bypassed in one RPS channel at a time.

Since a single failure neither causes nor prevents the protection system actuation, and protection channels are isolated from control channels through fiber optic cables, this arrangement meets the requirements of IEEE 279 (Reference 2.).

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The portions of the RPS instrumentation that develop signals and trips for variable overpower provide plant protection during certain AOOs, and assist the Engineered Safety Features (ESF) in the mitigation of certain accidents.

The Variable Overpower trip provides protection against core damage during the following events:

1. Steam Line Break
2. Uncontrolled CEA withdrawal from Low Power
3. Uncontrolled CEA withdrawal at Power
4. CEA Ejection

The Variable Overpower trip meets Criterion 3 for inclusion as a technical specification as it is part of a system on the primary success path for reactivity control.

LCOs

The LCO on the variable overpower trip ensures that the violation of the Safety Limits for the reactor core and RCS is prevented during normal operations and AOOs and assists the engineered safety features system during the CEA ejection accident.

The allowable values setpoints for ceiling and rate are selected large enough to prevent spurious trips during performance design base transients on reactor power cutback. The setpoint is low enough for the system to maintain a margin to unacceptable fuel cladding damage should a CEA withdrawal event, or rod ejection accident occur.

Only the allowable values setpoints are specified for each RPS trip function in the LCO. Each allowable value is specified such that the analytical limit assumed in the safety analysis is conservative including all applicable setpoint uncertainties.

(continued)

BASES

APPLICABILITY

The Variable Overpower trip is applicable in MODES 1 and 2 because the reactor can be critical in these modes. The trip is designed to take the reactor subcritical which assists (as described above) in mitigating the consequences of the particular accidents and AOOs listed.

In MODES 3, 4, and 5, the main concern is for a return to power event. The reactor is protected during this event by the High Log Power trip, and therefore, the above trip does not need to be OPERABLE.

ACTIONS

If a protection channel of a given process variable becomes inoperable, the goal shall be to return the inoperable channel to service as soon as practical, but no later than prior to returning to MODE 2 following entry into MODE 5. This approach to bypass/trip in four channel protection systems is consistent with the applicable criteria of IEEE standard 279.

The Unit Review Group (URG) shall determine the desirability of maintaining any channel in bypass pursuant to Administrative Controls.

A.1

RPS coincidence logic is normally 2/4. If the number of channels OPERABLE is one less than the Total Number of Channels (i.e., only 3 out of 4 available), startup or power operation is allowed to continue as long as the inoperable channel is placed in bypass or trip within 1 hour. The provision of 4 independent and redundant trip channels allows one channel to be bypassed (removed from service) during operations, placing the RPS in 2/3 coincidence logic. It is preferable to place an inoperable channel in bypass rather than trip, since no single additional failure can either spuriously trip the reactor, or prevent it from tripping.

If the channel fails (or is placed) in the tripped condition, just one spurious signal from any of the other three channels will cause the reactor to trip. Although this is acceptable from an accident analysis standpoint, it is not good operating practice. The time allotted to bypass or trip the channel allows the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

(continued)

BASES

ACTIONS (continued)

A.2

A possible source of a failed Variable Overpower trip channel is the failure of the associated ex-core detector channel. The ex-core detectors also provide input to the Core Protection Calculators (CPCs) which develop the departure from nucleate boiling ratio (DNBR) and local power density (LPD) parameters and trips. The average power level provides input to the Steam Generator Level-Low trip channels to calculate trip setpoints depending on plant power. The failure of a average power level channel may therefore affect its associated CPC channel, along with the DNBR and LPD trips, and Steam Generator Level - Low Trip. Thus, if one Variable Overpower trip channel is inoperable, it must be placed in bypass or trip, and the Steam Generator Level - Low trip must be placed in bypass or trip. The associated DNBR and LPD trip channels must be investigated and must also be placed in bypass or trip if they are affected by the loss of the Variable Overpower trip channel.

A.3

An inoperable channel must be returned to OPERABLE status prior to entry into MODE 2 following the next entry into MODE 5. This is in keeping with the general philosophy of not operating the plant in an off-normal condition indefinitely. The time allowed should be adequate to repair the inoperable channel. If the channel cannot be returned to OPERABLE status by the end of the next entry into MODE 5, the problem must be taken care of before startup and power operation can commence.

B.1

With the number of channels OPERABLE one less than the Minimum Channels Operable requirement, one inoperable channel must be placed in bypass, and the other channel must be placed in trip within the required Completion Time. With one channel of protective instrumentation bypassed, the RPS is in 2/3 logic, but with another channel failed, the RPS may be operating with a 2/2 logic (this assumes the channel failed in the non-conservative direction away from the trip setpoint). This is outside the assumptions made in the analyses and must be corrected. To correct the

(continued)

BASES

ACTIONS (continued)

problem, the second channel is placed in trip. This places the RPS in a 1/2 logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

Operation in MODES 1 and 2 may continue until the next CHANNEL FUNCTIONAL TEST. Operation in MODES 1 and 2 cannot continue beyond that time.

B.2

As explained above for action A.2, the loss of a Variable Overpower trip channel can cause the loss of the associated CPC channel and Steam Generator Level - low trip. For this reason, if a second Variable Overpower - High trip channel is inoperable, it must be placed in bypass or trip, and the associated Steam Generator Level - Low trip must be placed in bypass or trip. The associated DNBR and LPD trip channels must be investigated and must also be placed in bypass or trip if they are affected by the loss of the Variable Overpower trip channel.

C.1

If action A or B cannot be completed within the required completion time, the reactor must be brought to a MODE where the action statements do not apply. The six hours is adequate to shutdown the plant in a controlled manner, while keeping the risk of operating in this condition at an acceptable level.

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

Performing a CHANNEL CHECK once per 12 hours ensures that any channel which drifts beyond the normal expected instrument drift, as compared to other identical channels, is detected within a reasonable amount of time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Data Processing System (DPS) continuously performs a cross channel comparison and will initiate an alarm to warn operators that a channel has drifted out-of-tolerance or is not working properly.

The reason the CHANNEL CHECK is performed instead of a more thorough test, is that it imposes much less of a burden on operations than any of the other instrument channel tests. If the other instrument tests are performed at their appropriate frequencies, this test is very reliable.

SR 3.3.3.2

A daily calibration is performed when THERMAL POWER ≥ 15% RTP, and the Linear Power Level signals and the CPC addressable constant multipliers are adjusted to make the CPC Delta T Power and Nuclear Power calculations agree with the calorimetric calculation if the absolute difference is ≥ 2%. These checks (and if necessary, the adjustment of the Linear Power Level signals and the CPC addressable constant coefficients) are adequate to ensure that the accuracy of these CPC calculations (and the resultant DNBR and LPD trip setpoints) is maintained within the analyzed error margins with respect to reference indications.

SR 3.3.3.2 (continued)

The excore neutron detectors are excluded from the CHANNEL CALIBRATIONS because of their inaccessibility, and the difficulty of simulating a meaningful neutron flux signal. Also, operating experience has shown them to be very reliable.

SR 3.3.3.3

The three ex-core detectors in each channel are far enough from the core that they are exposed to flux from all heights in the core, although it is desired that they only read their particular level. The CPCs adjust for this flux overlap by using the predetermined shape annealing matrix elements in CPC software.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

For matrix elements to be meaningful, the individual ex-core subchannel gains must be adjusted every 31 days (with THERMAL POWER ≥ 15% RTP), using incore detectors to determine actual core axial power shape.

Operating experience has shown the specified interval to be adequate to ensure accurate instrument readings and input.

SR 3.3.3.4

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire channel will perform its intended function when needed. It is not necessary to perform this test more often because operating experience has shown that the instrumentation stays quite reliable within that time frame.

Major portions of the Reactor Protection system are monitored and/or tested by the automatic test network. Those portions of the system which are not amenable to automatic testing because they involve actuation of electromechanical devices, or involve devices which are not within the PPS cabinets, can be tested manually. The automatic test network is capable of performing tests during reactor operation. The automatic testing does not degrade the ability of the RPS to perform its intended function.

Detail description of the RPS automatic test features are described in CESSAR-DC Chapter 7.2.

SR 3.3.3.5

Performance of a CHANNEL CALIBRATION every 92 days ensures that the channels are reading accurately and within specified tolerances. Operating experience has shown this test interval to be satisfactory.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

[SR 3.3.3.6]

Performance of a CHANNEL CALIBRATION every refueling ensures that the channels are reading accurately and within specified tolerances. This CHANNEL CALIBRATION may require entry into containment for equipment check and adjustment. Since that cannot be done during the 92 day CHANNEL CALIBRATION (unless the reactor is shutdown), this test is performed every refueling. Operating experience has shown that this test interval is satisfactory.

REFERENCES

1. CESSAR-DC, chapters 7 "Instrumentation and Controls", and 15 "Accident Analysis".

2. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Generating Stations", April 5, 1972.

16A.6.4 B 3.3.4 LOG POWER LEVEL - HIGH

B 3.3 INSTRUMENTATION

B 3.3.4 Log Power Level - High

BASES

BACKGROUND	Logarithmic power level is measured by the ex-core detector safety channels, and processed through amplification and bistable processors to provide a Log Power Level - High trip for the reactor protection system (RPS). The signals are processed redundantly for each of the four RPS channels by separate logarithmic power level trip channels. A 2/3 trip logic is all that is required to provide adequate plant protection, since a failure of one channel still ensures a reactor trip would be generated by the two remaining OPERABLE channels. When performing maintenance, testing, or removing a failed channel from service, RPS logic for the Log Power Level - High trip is changed from 2/4 to 2/3 by trip channel bypassing. All RPS trips can be trip channel bypassed, providing each is bypassed in one RPS channel at a time. In addition to the trip channel bypass, there is also the 10^{-3}% RTP operating bypass for the Log Power Level - High trip. This bypass is enabled, manually in all four RPS channels, when plant conditions do not warrant the trip protection. The operating bypass is automatically removed when enabling bypass conditions are no longer satisfied. Since a single failure neither causes nor prevents the protection system actuation, and no protection channels are isolated from control channels through fiber cables, this arrangement meets the requirements of IEEE 279.
APPLICABLE SAFETY ANALYSIS	The Log Power Level - High trip protects the integrity of the fuel cladding and helps protect the Reactor Coolant System (RCS) pressure boundary in the event of a unplanned criticality from a shutdown condition (caused by a Control Element Assembly (CEA) withdrawal event), and alerts the operator (via alarm) to a boron dilution event.

(continued)

BASES

LCOs

The LCO on the Log Power Level - High trip ensures that violation of the Safety Limits for the reactor core and RCS is prevented during a continuous CEA withdrawal from low power levels event. Also, it ensures that the log power level channels are available to detect and alert the operator to a boron dilution event.

The allowable value setpoint is high enough to provide an operating envelope that prevents unnecessary Log Power Level - High reactor trips during normal plant operations. The setpoint is low enough for the system to maintain a margin to unacceptable fuel cladding damage should a CEA withdrawal event.

Only the Allowable Values are specified for each RPS trip function in the LCO. Each allowable value is specified such that the analytical limit assumed in the safety analysis is conservative including all applicable setpoint uncertainties.

APPLICABILITY

The Log Power Level - High trip is applicable in MODES 2, 3, 4, and 5 with the Reactor Trip Circuit Breakers (RTCBs) closed and power available to the CEA drive system. It is required for protection against CEA withdrawal events originating below 10^{-3}% RTP. For events originating above this power level, other RPS trips provide adequate protection.

In MODES 3, 4, and 5 with the RTCBs open, the CEAs are not capable of withdrawal and the Log Power Level - High trip does not have to be OPERABLE. However, two Log Power Level channels must be OPERABLE to ensure proper indication of neutron population, and to indicate a boron dilution event.

ACTIONS

If a protection channel of a given process variable becomes inoperable, the goal shall be to return the inoperable channel to service as soon as practical, but no later than prior to returning to MODE 4 following entry into MODE 5. This approach to bypass/trip in four channel protection systems is consistent with the applicable criteria of IEEE standards 279.

(continued)

BASES

ACTIONS (continued)

The Unit Review Group (URG) shall determine the desirability of maintaining any channel in bypass pursuant to Administrative Controls.

A.1

RPS coincidence logic is normally 2/4. If the number of channels OPERABLE is one less than the Total Number of Channels (i.e., only 3 out of 4 available) with the RTCBs closed, operation is allowed to continue as long as the inoperable channel is placed in bypass or trip within 1 hour. The provision of 4 independent and redundant trip channels allows one channel to be bypassed (removed from service) during operations, placing the RPS in 2/3 coincidence logic. It is preferable to place an inoperable channel in bypass rather than trip, since no single additional failure can either spuriously trip the reactor, or prevent it from tripping.

If the channel fails (or is placed) in the tripped condition, just one spurious signal from any of the other three channels will cause the reactor to trip. Although this is acceptable from an accident analysis standpoint, it is not good operating practice. The time allotted to bypass or trip the channel allows the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

A.2

An inoperable channel must be returned to OPERABLE status prior to entry into MODE 4 following the next entry into MODE 5. This is in keeping with the general philosophy of not operating the plant in an off-normal condition indefinitely. The time allowed should be adequate to repair the inoperable channel. If the channel cannot be returned to OPERABLE status by the end of the next entry into MODE 5, the problem must be taken care of before startup and power operation can commence.

(continued)

BASES

ACTIONS (continued)

B.1

With the number of channels OPERABLE one less than the Minimum Channels Operable requirement in MODE 2, one inoperable channel must be placed in bypass, and the other channel must be placed in trip within the required Completion Time. With one channel of protective instrumentation bypassed, the RPS is in 2/3 logic, but with another channel failed, the RPS may be operating with a 2/2 logic (this assumes the channel failed in the non-conservative direction away from the trip setpoint). This is outside the assumptions made in the analyses and must be corrected. To correct the problem, the second channel is placed in trip. This places the RPS in a 1/2 logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

Operation in MODE 2 may continue until the next CHANNEL FUNCTIONAL TEST. Operation in MODE 2 cannot continue beyond that time.

C.1

If action A or B cannot be completed within the required completion time, the reactor must be brought to a MODE where the action statements do not apply. The six hours is adequate to shutdown the plant in a controlled manner, while keeping the risk of operating in this condition at an acceptable level.

D.1

If the number of channels OPERABLE is one less than the Minimum Channels Operable requirement of the table while operating in MODES 3, 4, or 5 with the RTCBs closed, one channel must be placed in bypass, and the other placed in trip within 1 hour, for the same reasons as B.1. The reasons for the allotted time to perform the Required Actions are the same as B.1.

(continued)

BASES

ACTIONS (continued)

D.2

In MODES 3, 4, and 5, with the RTCBs closed, if the number of channels OPERABLE becomes one less than the Minimum Channels OPERABLE requirement of the table, in addition to the Required Actions of D.1, one channel must be restored to OPERABLE status within 48 hours. The reason the plant is not allowed to operate until the next CHANNEL FUNCTIONAL TEST is because the Log Power Level - High trip is the only trip which will protect the plant against an uncontrolled CEA withdrawal event from MODES 3, 4, or 5. The allowed time is adequate to restore one channel to OPERABLE status while keeping the risk of operating in this condition at an acceptable level.

E.1

If the Required Actions of condition D. are not met within the required Completion Time, the RTCBs must be open to preclude the possibility of a Continuous CEA withdrawal event. The 1 hour Completion Time is adequate to take the Required Action while keeping the risk of operating in this condition at an acceptable level.

F.1

If the number of channels OPERABLE is one less than the Minimum Channels Operable requirement of the table in MODES 3, 4, or 5 with the RTCBs open, all operations involving positive reactivity additions must be suspended immediately. This is because there is only one channel left for reactor power indication, and single failure criteria cannot be maintained. The risk of allowing positive reactivity additions with only one channel of indication OPERABLE is unacceptable, so the additions must be suspended.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

Performing a CHANNEL CHECK every 12 hours ensures that any channel which drifts beyond the normal expected instrument drift, as compared to other identical channels, is detected within a reasonable amount of time.

The Data Processing System (DPS) continuously performs a cross channel comparison and will initiate an alarm to warn operators that a channel has drifted out-of-tolerance or is not working properly.

The reason the CHANNEL CHECK is performed instead of a more thorough test, is that it imposes much less of a burden on operations than any of the other instrument channel tests. If the other instrument tests are performed at their appropriate frequencies, this test is very reliable.

SR 3.3.4.2 and 3.3.4.3

A CHANNEL FUNCTIONAL TEST is performed every 92 days, and prior to each startup if not performed within the last 7 days, to ensure the entire channel will perform its intended function when needed. It is not necessary to perform this test more often because operating experience has shown that the instrumentation stays reliable within that time frame.

Major portions of the Reactor Protection System are monitored and/or tested by the automatic test network. Those portions of the system which are not amenable to automatic testing because they involve actuation of electromechanicals devices, or involve devices which are not within the PPS cabinets, can be tested manually. The automatic test network is capable of performing tests during reactor operation. The automatic testing does not degrade the ability of the RPS to perform its intended function.

Detail description of the RPS automatic test features are described in CESSAR-DC Chapter 7.2.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.4

Performance of a CHANNEL CALIBRATION every 18 months ensures that the channels are reading accurately and within specified tolerances. Operating experience has shown this test interval to be satisfactory.

The excore neutron detectors are excluded from the CHANNEL CALIBRATIONS because of their inaccessibility, and the difficulty of simulating a meaningful neutron flux signal. Operating experience has shown them to be reliable.

SR 3.3.4.5

Performance of a CHANNEL CALIBRATION every 18 months ensures that the operating bypass is operating accurately and within the specified tolerances. Operating experience has shown this test interval to be satisfactory.

SR 3.3.4.6

Verifying the logic for the operating bypass to be OPERABLE within the specified time interval, ensures that the bypass and its permissive setpoint function as designed prior to entering a MODE where they are required. Operating experience has shown the specified interval to be satisfactory.

REFERENCES

1. CESSAR-DC, Chapters 7 "Instrumentation and Controls", and 15 "Accident Analysis".

2. IEEE Standard 279-1971, Criteria for Protection Systems for Nuclear Power Generating Stations", April 5, 1972.

16A.6.5 B 3.3.5 CORE PROTECTION CALCULATORS (CPCs)

B 3.3 INSTRUMENTATION

B 3.3.5 Core Protection Calculators (CPCs)

BASES

BACKGROUND

The CPCs perform the calculations required to derive the departure from nucleate boiling ratio (DNBR) and local power density (LPD) parameters, and their associated reactor protective system (RPS) trips. Four separate CPCs perform the calculations independently for each of the four RPS channels. The CPCs provide outputs to drive display indications (DNBR margin, LPD margin, and calibrated neutron flux power levels) and provide low DNBR and high LPD pretrip and trip signals. The CPC channel outputs for the DNBR Low and LPD High trips are provided to four Bistable processors in the form of contact on or off to provide the designed 2/4 logic.

Each CPC receives the following inputs:

1. Hot leg and cold leg temperatures

2. Pressurizer pressure

3. Reactor coolant pump speed

4. Ex-core neutron flux levels

5. Target control element assembly (CEA) positions

6. Control Element Assembly Calculator (CEAC) penalty factors

Each CPC is programmed with "addressable constants". These are various alignment values, correction factors, etc. that are required for the DNBR and LPD computations. They can be accessed for display, or for the purpose of changing them as necessary during on-line processing.

The CPCs use this constant and variable information to perform a number of calculations. These include the calculation of CEA group and subgroup deviations (and the assignment of conservative "penalty factors"); correction and calculation of average axial power distribution (based on ex-core flux

(continued)

BASES

BACKGROUND (continued)

levels and CEA positions); calculation of coolant flow (based on pump speed); and calculation of calibrated average power level (based on ex-core flux levels and delta-T power).

The DNBR calculation considers primary pressure, inlet temperature, coolant flow, average power, axial power distribution, radial peaking factors, and CEA deviation penalty factors from the CEACS, to calculate the state of the limiting (hot) coolant channel in the core. A DNBR Low trip occurs when the calculated value reaches the minimum DNBR trip setpoint.

The LPD calculation considers axial power distribution, average power, and radial peaking factors (based upon target CEA position) and CEAC penalty factors to calculate the current value of compensated peak power density. A LPD High trip occurs when the calculated value reaches the trip setpoint.

The 4 CPC channels provide input to the 4 DNBR Low and 4 LPD High RPS trip channels. They effectively act as the sensor (using many inputs) for these trips. A 2/3 trip logic is all that is required to provide adequate plant protection, since a failure of one channel still ensures a reactor trip would be generated by the two remaining OPERABLE channels. When performing maintenance, testing, or removing a failed CPC channel from service, RPS logic for the DNBR and LPD trips is changed from 2/4 to 2/3 by trip channel bypassing. All RPS trips can be trip channel bypassed, providing each is bypassed in one RPS channel at a time.

In addition to the trip channel bypass, there is also the operating bypass for the CPC channels. This bypass is enabled manually in all four CPC channels when RTP is below ~~1%~~ power. The bypass effectively removes the DNBR and LPD trips from the RPS logic circuitry. The operating bypass is automatically removed when RTP is above ~~1%~~.

the specified Thermal Power level

Since a single failure neither causes nor prevents the protection system actuation, and protection channels are isolated from control channels through fiber optic cables, this arrangement meets the requirements of IEEE 279 (Reference 2.).

(continued)

BASES

APPLICABLE SAFETY ANALYSIS

The CPCs perform the calculations required to derive the departure from nucleate boiling ratio (DNBR) and local power density (LPD) parameters, and their associated reactor protective system (RPS) trips. The DNBR and LPD trips provide plant protection during the Anticipated Operational Occurrences (AOOs) stated in the bases for LCO 3.2.2, and assist the Engineered Safety Features (ESF) systems in the mitigation of the accidents listed in that bases.

LCOs

The LCO on the CPCs ensures that the following requirements are met:

1. A reactor trip will be initiated when necessary.
2. The minimum (2/3) coincidence logic required for RPS instrumentation is maintained.
3. Sufficient redundancy exists to permit the removal of one CPC channel from service for testing or maintenance, while still meeting the 2/3 logic requirement.

APPLICABILITY

Because the CPCs provide the inputs to the DNBR Low and LPD High trips, they are required to be OPERABLE in the same MODES, and for the same reasons, as those trips.

~~In MODES 3, 4, and 5, the main concern is for a return to power event. The reactor is protected during this event by the High Log Power trip. Therefore, the CPCs do not need to be OPERABLE.~~

ACTIONS

If a protection channel of a given process variable becomes inoperable, the goal shall be to return the inoperable channel to service as soon as practical, but no later than prior to returning to MODE 2 following entry into MODE 5. This approach to bypass/trip in four channel protection systems is consistent with the applicable criteria of IEEE standards 279.

The Unit Review Group (URG) shall determine the desirability of maintaining any channel in bypass pursuant to Administrative Controls.

(continued)

BASES

ACTIONS (continued)

A.1

RPS coincidence logic is normally 2/4. If the number of CPC channels OPERABLE is one less than the Total Number of Channels (i.e., only 3 out of 4 available), startup or power operation is allowed to continue as long as the DNBR and LPD channels associated with the inoperable CPC are placed in bypass or trip within 1 hour. The provision of 4 independent and redundant DNBR and LPD trip channels allows one channel to be bypassed (removed from service) during operations, placing the RPS in 2/3 coincidence logic. It is preferable to place an inoperable channel in bypass rather than trip, since no single additional failure can either spuriously trip the reactor, or prevent it from tripping.

If the channel fails (or is placed) in trip, just one spurious signal from any of the other three channels will cause the reactor to trip. Although this is acceptable from an accident analysis standpoint, it is not good operating practice. The time allotted to bypass or trip the channel allows the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

A.2

An inoperable channel must be returned to OPERABLE status prior to entry into MODE 2 following the next entry into MODE 5. This is in keeping with the general philosophy of not operating the plant in an off-normal condition indefinitely. The time allowed should be adequate to repair the inoperable channel. If the channel cannot be returned to OPERABLE status by the end of the next entry into MODE 5, the problem must be taken care of before startup and power operation can commence.

B.1

With the number of CPC channels OPERABLE one less than the Minimum Channels Operable requirement, the DNBR and LPD channels associated with one inoperable CPC must be placed in bypass, and the DNBR and LPD channels associated with the other inoperable CPC must be placed in trip within the required Completion Time. With one channel of protective instrumentation bypassed, the RPS is in 2/3 logic, but with another channel

(continued)

ACTIONS (continued)

failed, the RPS may be operating with a 2/2 logic (this assumes the channel failed in the non-conservative direction away from the trip setpoint). This is outside the assumptions made in the analyses and must be corrected. To correct the problem, the second channel is placed in trip. This places the RPS in a 1/2 logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

Operation in MODES 1 and 2 may continue until the next CHANNEL FUNCTIONAL TEST. Operation in MODES 1 and 2 cannot continue beyond that time.

C.1

If a non-bypassed CPC has more than 3 auto restarts within a 12 hour interval, it may not be completely reliable. Therefore, a CHANNEL FUNCTIONAL TEST must be performed on the CPC to ensure it is functioning properly. The 24 hours is adequate to perform the test while still keeping the risk of operating in this condition to acceptable levels.

D.1

If action A, B, or C cannot be completed within the required completion time, the reactor must be brought to a MODE where the action statements do not apply. The six hours is adequate to shutdown the plant in a controlled manner, while keeping the risk of operating in this condition at an acceptable level.

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1

Performing a CHANNEL CHECK every 12 hours ensures that any CPC that is not reading the same (within allowed tolerances) as the other OPERABLE CPCs, is detected within a reasonable length of time. It also serves as a check on the sensors feeding each CPC.

The Data Processing System (DPS) continuously performs a cross channel comparison and will initiate an alarm to warn operators that a channel has drifted out-of-tolerance or is not working properly.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

The CHANNEL CHECK imposes much less of a burden on operations than other more thorough tests, and operating experience has shown that it is adequate as long as the other instrument tests are performed at their specified frequencies.

SR 3.3.5.2

The CPC auto restart count is checked every 12 hours to monitor the CPC for normal operation. If 3 or more auto restarts of a non-bypassed CPC occurs within a 12 hour period, the CPC may not be completely reliable. Therefore, the Required Action of Condition C. must be performed.

SR 3.3.5.3

If a CPC cabinet High Temperature alarm is received, it is possible for the CPC to be affected and not be completely reliable. Therefore, a CHANNEL FUNCTIONAL TEST must be performed within 12 hours. The interval is based on operating experience and is adequate to ensure CPC operability.

SR 3.3.5.4

The RCS flow rate indicated by each CPC is verified to be less than or equal to the actual RCS total flow rate every 12 hours with THERMAL POWER $\geq$ 70% RTP. This check (and if necessary, the adjustment of the CPC addressable flow constant coefficients) ensures that the DNBR setpoint is conservatively adjusted with respect to actual flow indications [as determined by the Core Operating Limits Supervisory System (COLSS)].

SR 3.3.5.5

A daily calibration (heat balance) is performed when THERMAL POWER is $\geq$ 15%. The Linear Power Level signal and the CPC addressable constant multipliers are adjusted to make the CPC Delta T Power and Nuclear Power calculations agree with the calorimetric calculation if the absolute difference is $\geq$ 2%. These checks (and if necessary, the adjustment of the Linear Power Level signal and the CPC addressable constant coefficients) are adequate to ensure that the accuracy of these CPC calculations is maintained within the analyzed error margins.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

The excore neutron detectors are excluded from the daily calibrations because of their inaccessibility, and the difficulty of simulating a meaningful neutron flux signal. Operating experience has shown them to be very reliable.

SR 3.3.5.6

The RCS flow rate indicated by each CPC is verified to be less than or equal to the actual RCS total flow rate every 31 days. This check (and if necessary, the adjustment of the CPC addressable flow constant coefficients) ensures that the DNBR setpoint is conservatively adjusted with respect to actual flow indications as determined by a calorimetric calculation (the most accurate means of determining RCS flowrate). Operating experience has shown the specified interval to be adequate.

SR 3.3.5.7

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire channel will perform its intended function when needed. It is not necessary to perform this test more often because operating experience has shown that the instrumentation stays quite reliable within that time frame.

Major portions of the Reactor Protection System are monitored and/or tested by the automatic test network. Those portions of the system which are not amenable to automatic testing because they involve actuation of electromechanical devices, or involve devices which are not within the PPS cabinets, can be tested manually. The automatic test network is capable of performing tests during reactor operation. The automatic testing does not degrade the ability of the RPS to perform its intended function.

Detail description of the SPR automatic test features are described in CESSAR-DC Chapter 7.2.

SR 3.3.5.8

Performance of a CHANNEL CALIBRATION every refueling ensures that the operating bypasses are operating accurately and within the specified tolerances. Operating experience has shown this test interval to be satisfactory.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.9

A CHANNEL FUNCTIONAL TEST including the injection of simulated process signals as close to the sensors as possible is performed every eighteen months to ensure that the entire RPS trip channel, including the signal paths between the detectors and the RPS instrumentation, are able to perform their intended function. The test requires the reactor to be shutdown. More frequent performance of this test is not necessary because operating experience has shown the equipment to be reliable within this time frame.

SR 3.3.5.10

Verifying the logic for the operating bypasses to be OPERABLE within the specified time interval, ensures that the bypasses and their permissive setpoints function as designed prior to entering a MODE where they are required. Operating experience has shown the specified interval to be satisfactory.

SR 3.3.5.11

The three ex-core detectors in each channel are far enough from the core to be exposed to flux from all heights in the core, although it is desired that they only read their particular level. The CPCs adjust for this flux overlap by using the predetermined shape annealing matrix elements in the CPC software.

After refueling, it is necessary to reestablish the shape annealing matrix elements for the ex-core detectors based on more accurate incore detector readings. This is necessary because refueling could possibly produce a significant change in the of the flux fields that the ex-core detectors are reading.

Incore detectors are inaccurate at low power levels. THERMAL POWER should be significant, but less than 70% to perform an accurate axial shape calculation used to derive the shape annealing matrix elements.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	By restricting power to ≤ 70% until shape annealing matrix elements are verified, excessive local power peaks within the fuel are avoided. Operating experience has shown this interval to be acceptable.

REFERENCES

1. CESSAR-DC, chapters 7 "Instrumentation and Controls", and 15 "Accident Analysis".

2. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Generating Stations", April 5, 1972.

3. ~~Shutdown~~ Cessar-DC, Section 19.8 "Shutdown Risk Report"

Amendment I
December 21, 1990

16A.6.6 B 3.3.6 CONTROL ELEMENT ASSEMBLY CALCULATORS (CEACs)

B 3.3 INSTRUMENTATION

B 3.3.6 Control Element Assembly Calculators (CEACs)

BASES

BACKGROUND	The CEACs perform the calculations required to determine the position of Control Element Assemblies (CEAs) within their subgroups for the reactor protection system (RPS). Two independent CEACs compare the position of each CEA to its subgroup position. If a deviation is detected by either CEAC, an annunciator sounds, and appropriate "penalty factors" are transmitted to all Core Protection Calculators (CPCs). These penalty factors conservatively adjust the effective operating margins to the low Departure from Nucleate Boiling Ratio (DNBR) and high Local Power Density (LPD) trips. Each CEAC also drives a display of individual CEA positions in bar chart format and current values of the penalty factors on channel B and C Operator's Module. Each CEA has two separate reed switch assemblies that measure CEA position independently for each CEAC.
APPLICABLE SAFETY ANALYSIS	The effect of any misoperated CEA on the core power distribution will be assessed by the CEACs, and an appropriately augmented power distribution penalty factor will be supplied as input to the CPCs. As the reactor core responds to the reactivity changes caused by the misoperated CEA, and the ensuing reactor coolant and doppler feedback effects, the CPCs will initiate a DNBR - Low, or LPD - High trip signal, if specified acceptable fuel design limits (SAFDLs) are approached. Therefore, although the CEACs do not provide a direct reactor trip function, their input to the CPCs is taken credit for in the CEA misoperation analysis.

(continued)

BASES

LCOs	The LCO ensures that the CEACs are available for input to the CPCs and that the following requirements are met: 1. Acceptable power distribution limits are maintained 2. The potential effects of CEA misalignments are limited to acceptable levels. 3. Sufficient redundancy exists to remove a channel from service for maintenance or testing.
APPLICABILITY	The CEACs provide input to the CPCs, which are required to be OPERABLE in MODES 1 and 2 for the reasons stated in the bases for LCO 3.3.5. Therefore, the CEACs are required to be OPERABLE in the same MODES as the CPCs.
ACTIONS	<u>A.1</u> With one CEAC inoperable, the second CEAC still provides a comprehensive set of comparison checks on individual CEAs within subgroups, as well as outputs to all CPCs, CEA deviation alarms, and position indications for display. Verification that each CEA is within [7] inches of other CEAs in its group within 4 hours provides a check on the proper position of all CEAs, and operation of the remaining CEAC. The 4 hours is adequate based upon engineering judgment and operating experience. <u>A.2</u> If a CEAC becomes inoperable, the goal shall be to return the channel to service as soon as practical, but no later than 7 days. Engineering judgment and operating experience indicate that the allowed time is sufficient to correct any such deficiency.

(continued)

BASES

ACTIONS (continued)

NOTE

All of the Required Actions for condition B. are taken as if both CEACs are inoperable.

B.1

If required action A is not met or both CEACs are inoperable, meeting the DNBR margin requirements of LCO 3.2.4 ensures that power level and axial shape index (ASI) are within a conservative region of operation based on actual core conditions.

In addition to the above actions, the Reactor Power Cutback (RPCB) system must be disabled. This ensures that CEA position will not be affected by RPCB operation. The CPC's cannot recognize an RPCB without the CEAC in operation.

B.2

When both CEACs are inoperable, or required action A is not met, the "full out" CEA reed switches provide acceptable indication of CEA position. Therefore, the CEAs will remain fully withdrawn, except as required for specified testing or axial shape control. This verification ensures that undesired perturbations in local fuel burnup are prevented.

B.3

When both CEACs are inoperable, or required action A is not met, the "RSPT/CEAC Inoperable" addressable constant in each of the CPCs is set to indicate that both CEACs are inoperable. This provides a conservative penalty factor to ensure that a conservative effective margin is maintained by the CPCs in the computation of DNBR and LPD trips.

B.4

With both CEACs inoperable, the Control Element Drive Mechanism Control System (CEDMCS) is placed and maintained in "standby" to prevent inadvertent motion (and possible misalignment) of the CEAs.

(continued)

BASES

ACTIONS (continued)

B.5

With both CEACs inoperable, a comprehensive set of comparison checks on individual CEAs within groups must be made within 4 hours. Verification that each CEA is within [7] inches of other CEAs in its group provides a check that no CEA has deviated from its proper position within the group.

C.1

If a non-bypassed CEAC has more than 3 auto restarts, within a 12 hour period, it may not be completely reliable. Therefore, a CHANNEL FUNCTIONAL TEST must be performed on the CEAC to ensure it is functioning properly. The 24 hours is adequate to perform the test while still keeping the risk of operating in this condition at an acceptable level.

D.1

If action B or C cannot be completed within the required completion time, the reactor must be brought to a MODE where the action statements do not apply. The six hours is adequate to shutdown the plant in a controlled manner, while keeping the risk of operating in this condition at an acceptable level.

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1

Verifying the position of each CEA to be within [ ] inches of all other CEAs in its group every 4 hours whenever any CEAC is inoperable, ensures that no individual CEA is out of alignment.

SR 3.3.6.2

Performing a CHANNEL CHECK every 12 hours ensures that an inoperable CEAC is detected within a reasonable length of time. It also serves as a check on the sensors feeding each CEAC.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Data Processing System (DPS) continuously performs a cross channel comparison and will initiate an alarm to warn operators that a channel has drifted out-of-tolerance or is not working properly.

The CHANNEL CHECK imposes much less of a burden on operations than other more thorough tests, and operating experience has shown that it is adequate as long as the other tests are performed at their specified frequencies.

SR 3.3.6.3

The CEAC auto restart count is checked every 12 hours interval to monitor the CEACs for normal operation. If 3 or more auto restarts of a CEAC occurs within a 12 hour period, the CEAC may not be completely reliable. Therefore, the Required Action of Condition C. must be performed.

SR 3.3.6.4

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire channel will perform its intended function when needed. It is not necessary to perform this test more often because operating experience has shown that the instrumentation stays reliable within that time frame.

Major portions of the Reactor Protection System are monitored and/or tested by the automatic test network. Those portions of the system which are not amenable to automatic testing because they involve actuation of electromechanical devices, or involve devices which are not within the PPS cabinets, can be tested manually. The automatic test network is capable of performing tests during reactor operation. The automatic testing does not degrade the ability of the RPS to perform its intended function.

Detail description of the RPS automatic test features are described in CESSAR-DC Chapter 7.2.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.5

A CHANNEL CALIBRATION is performed every refueling to ensure the entire channel will perform its intended function when needed and any adjustments to the circuitry are made as appropriate. It is not necessary to perform this test more often because operating experience has shown that the instrumentation stays reliable within that time frame.

SR 3.3.6.6

A CHANNEL FUNCTIONAL TEST including the injection of simulated process signals as close to the sensors as possible is performed every refueling to ensure that the entire CEAC channel, including the signal paths between the CEA RSPTs and the CPCs, are able to perform their intended function. More frequent performance of this test is not necessary because operating experience has shown the equipment to be reliable within this time frame.

REFERENCES

1. CESSAR-DC, chapters 7 "Instrumentation and Controls", and 15 "Accident Analysis".

2. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Generating Stations", April 5, 1972.

16A.6.7 B 3.3.7 REACTOR PROTECTIVE SYSTEM (RPS) LOGIC

B 3.3 INSTRUMENTATION

B 3.3.7 Reactor Protective System (RPS) Logic

BASES

BACKGROUND

Reactor Protective System (RPS) initiates a reactor trip to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary limits during Anticipated Operational Occurrences (AOOs), and assists the Engineered Safety Features (ESF) Systems in mitigating the consequences of accidents.

Four measurement channels with electrical and physical separation are provided for each parameter used in the direct generation of trip signals, with the exception of the control element assembly (CEA) position indication used in the Control Element Assembly Calculators (CEACs). When any two channels of like instrumentation receive a trip signal, a reactor trip is generated.

The Reactor Protection System (RPS) employs a logic scheme that provides a reactor trip when the bistables of any 2 of the 4 channels sensing the same input parameter trip. This is called a 2 out of four (2/4) trip logic.

It is possible to change the 2/4 logic to 2/3 logic in one channel at a time for any given input parameter by trip channel bypassing selected portions of the RPS logic.

The Reactor Protection System Logic can be subdivided into subsections: Local Coincidence Logic and Initiation Logic.

The Local Coincidence Logic determines if a coincidence exists in the tripping of like bistable logics (those monitoring the same parameter) in two or more channels. The actual bistables logics and upstream instrumentation are addressed in LCOs 3.3.1, 3.3.2, 3.3.3., 3.3.4, 3.3.5, and 3.3.6. If a coincidence occurs in two or more channels, the Initiation Logic is deenergized, resulting in a reactor trip.

(continued)

BASES

BACKGROUND (continued)

There are four Coincidence Logic Channels, comparing the outputs of all four channels of bistables, taken two, three and four at a time. A Coincidence Logic is formed by interrotating the bistable logic outputs in all four channels such that like bistables must be tripped in more than two channels to deenergize a Coincidence Logic Channel. Deenergizing more than two or selective two coincidence Logic Channels will deenergize associated Initiation Logic Channels, which open associated Reactor Trip Circuit Breakers (RTCBs).

The initiation logic consists of an OR circuit. The outputs from the associated Local Coincidence Logic for each parameters are provided to or circuit. The initiation circuit contains a time delay for noise filtering. The Initiation Logic generates two outputs, each initiating under voltage relay and shunt trip relays respectively.

There are four Initiation Logic Channels, each responsible for opening one RTCB (referred to as channels of RTCBs in Specification 3.3.5), if the associated coincidence logics deenergize.

Since a single failure neither causes nor prevents the protection system actuation, and no protection channels feed control channels, this arrangement meets the requirements of IEEE 279 (Reference 2.).

APPLICABLE SAFETY ANALYSIS

The RPS logic provides for automatic trip initiation to maintain the core, and RCS Pressure, Safety Limits during AOOs and assist the ESF systems in the mitigation of accidents. All transients and accidents that call for a reactor trip assume the RPS Logic is functioning as designed.

LCO

The LCO on the RPS logic channels ensures that each of the following requirements are met:

1. A reactor trip will be initiated when necessary.

2. The required protection system coincidence logic is maintained (minimum 2/3 logic, normal 2/4 logic).

(continued)

BASES

LCO (continued)

3. Sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance.

APPLICABILITY

The RPS Logic Channels are required to be operable in any MODE when the CEAs are capable of being withdrawn off the bottom of the core (i.e. RTCBs closed and power available to the CEDMs). This ensures the reactor can be tripped when necessary, but allows for maintenance and testing when the reactor trip is not needed.

ACTIONS

Normal RPS logic is 2/4. In case of trip channel bypass of any input parameter in accordance with Specification 3.3.1, 3.3.2, or 3.3.3, the Local Coincidence Logics are rendered into 2/3 logic appropriately.

Failures of individual bistable logics are addressed in LCOs 3.3.1, 3.3.2, 3.3.3, and 3.3.4. This specification addresses failures of the Local Coincidence Logic not addressed in the above, such as the failure of Matrix relay power supplies. This approach to bypass/trip in a four channel protection system is consistent with the applicable criteria of IEEE standard 279.

A.1

For the definition and diagram of a trip leg, see the bases for LCO 3.3.8 (Reactor Trip Circuit Breakers).

In MODES 1 and 2, if one or both Local Coincidence topic channels affecting the same trip leg are inoperable, the outputs from inoperable LCLs are not available for Initiation Logic channels. This result in the same effect as inoperable Initiation Logic channels requiring the same actions.

If one or both Local Coincidence Logic Channels (or Initiation Logic Channels) affecting the same trip leg are inoperable, operation may continue provided the affected RTCBs are opened within 1 hour. Opening the RTCBs eliminates the need for the initiation circuit. Allowing both outputs from

(continued)

BASES

ACTIONS (continued)

coincidence logics or initiation circuits (or coincidence Logic Channel) on the same trip leg to be Inoperable permits repair or replacement of failed power supply or associated wiring in accordance with the provisions of Action A.1.

The allotted completion time is adequate to open the affected [RTCBs] while maintaining risk of keeping them closed at an acceptable level.

A.2

If the RTCBs are not opened within 1 hour, the reactor must be brought to a con/ ..ion where the LCO does not apply. The 6 hours is reasonable to shutdown the reactor without initiating a transient requiring a reactor trip while maintaining risk at an acceptable level.

With only one Initiation Logic Channel (or Local Coincidence Logic channel) Inoperable, it is permissible to close the RTCB(s) in the affected trip leg for up to one hour while performing CHANNEL FUNCTIONAL TESTS on the RTCBs in the other trip leg. Without this provision, it would be impossible to open RTCBs during testing on the other trip leg without causing a reactor trip.

This is not permitted if both Initiation Logic Channels (or both Local Coincidence Logic Channels or combination of Local Coincidence Logic Channels and Initiation circuits resulting in both RTCBs in the same trip leg inoperable) are Inoperable on the same trip leg because an automatic reactor trip would be disabled.

The allotted time is adequate to perform the required RTCB testing, while maintaining risk of having the breakers closed at an acceptable leve..

B.1

In MODES 3, 4, and 5, if one or both Initiation Logic Channels (one or both Local coincidence Logic Channels) affecting the same trip leg are inoperable, the affected RTCBs must be opened within 1 hour. Opening the RTCBs eliminates the need for outputs from Local Coincidence Logic channels or the

(continued)

BASES

ACTIONS (continued)

initiation circuit. The 1 hour gives the operator time to open the RTCBs while maintaining the risk of not having the OPERABLE initiation logic at an acceptable level.

B.2.1

The inoperable channels must be restored to OPERABLE status within 48 hours. The time allotted should be adequate to repair the failed channels.

B.2.2

If the inoperable channels cannot be restored to OPERABLE status within 48 hours the RTCBs must be opened. This brings the plant to a condition where the LCO does not apply.

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire channel will perform its intended function. Operating experience has demonstrated that the digital circuits remains reliable within this time frame. It is not desirable to perform this test more often since there is always the possibility of receiving a spurious RPS trip during the test.

SR 3.3.7.2

Verifying the logic for trip channel bypasses OPERABLE within 92 days prior to each reactor startup, ensures that the bypasses function as designed prior to entering a MODE where they are required. Operating experience has shown this interval to be satisfactory.

The inability to trip channel bypass could adversely affect RPS reliability, since without this capability the RPS may be placed in 1/3 logic during channel testing or maintenance. This increases the likelihood of an inadvertent RPS actuation.

(continued)

BASES

REFERENCES

1. CESSAR-DC, Chapters 7 "Instrumentation and Controls", and 15 "Accident Analysis".

2. IEEE Standard 279-1971, Criteria for Protection Systems for Nuclear Power Generating Stations", April 5, 1972.

16A.6.8 B 3.3.8 REACTOR TRIP CIRCUIT BREAKERS (RTCBs)

B 3.3 INSTRUMENTATION

B 3.3.8 Reactor Trip Circuit Breakers (RTCBs)

BASES

BACKGROUND

The Reactor Trip Switchgear consists of 4 RTCBs which are operated as four channels. Power input to the reactor trip switchgear comes from two full-capacity motor generator (MG) sets (operated in parallel), such that the loss of either MG set does not deenergize the Control Element Drive Mechanisms (CEDMs). There are two separate CEDM power supply busses. Both Power supply Bus No. 1 and No. 2 are tied together through Synchronizer. Power is supplied from the MG sets to CEDM power bus via two redundant paths (trip legs). This ensures that a fault, or opening of a breaker in one trip leg (i.e., for testing purposes) will not interrupt power to the CEDM buses.

Each of the two trip legs consists of two RTCBs in series. Each RTCB is assigned to one Manual Reactor Trip Push button and one RPS Initiation Logic Circuit.

Thus each Trip circuit breaker is operated by either a Manual Reactor Trip pushbutton, or a Reactor Protection System (RPS) actuated initiation relays (automatic reactor trip).

Initiation Logic circuits in RPS actuates two initiation relay outputs. One relay output deenergize the undervoltage trip circuit, the other energizing the shunt trip circuit. This configuration gives redundancy and diversity.

During an automatic reactor trip, all initiation relays are deenergized, thereby opening all four breakers, and allowing the CEAs to fall into the core.

When a Manual Reactor Trip is initiated via the pushbuttons in the Control Room, the RPS trip path logic and the initiation relays are bypassed, and the signal is sent directly to the RTCBs.

(continued)

Amendment I
December 21, 1990

BASES

BACKGROUND (continued)

Each Manual Reactor Trip pushbutton operates a single breaker. Therefore, at least two pushbuttons must be depressed to cause a reactor trip. To ensure a reactor trip can be manually actuated with a single random failure in one breaker or its trip circuit, both sets of pushbuttons are required to be operable.

Each channel of RTCBs starts at the contacts which are actuated by the initiation relay, and the contacts which are actuated by the Manual Reactor Trip, for each breaker. Upstream of the initiation relay actuated contacts, the circuitry is considered to be RPS logic and falls under the requirements of LCO 3.3.7. Upstream of the Manual Reactor Trip contacts, the circuitry is considered to be Manual Reactor Trip circuitry and falls under the requirements of LCO 3.3.9.

Without reliable reactor trip circuit breakers and associated support circuitry, a reactor trip cannot occur when either initiated automatically, or manually. Therefore this LCO addresses the reactor trip circuit breakers and meets the requirements of GDC-21 and GDC-23.

APPLICABLE SAFETY ANALYSES

All of the transient and accident analyses which call for a reactor trip, assume that the RTCBs operate and interrupt power to the CEDMs.

LCOs

The LCO requires four RTCB channels (consisting of 1 breaker per channel) to be OPERABLE . This ensures that power is interrupted to CEDMs to fall into the core when any selective two out of four coincidence exists in trip signals.

APPLICABILITY

The RTCBs are required to be OPERABLE in any MODE when the CEAs are capable of being withdrawn off the bottom of the core (i.e., RTCBs closed and power available to the CEDMs). This ensures the reactor can be tripped when necessary, but allows for maintenance and testing when the reactor trip is not needed.

(continued)

Amendment I
December 21, 1990

BASES

ACTIONS

A.1

In MODES 1 and 2, if one channel becomes inoperable, operation may continue providing the affected breaker in that channel are opened within the 1 hour. This action is conservative since opening either RTCB on the other trip leg will cause a reactor trip.

The allotted Completion Time is adequate to open the affected RTCBs, while maintaining the risk of having them closed at an acceptable level.

A.2

If the breakers in the inoperable channel cannot be opened, the reactor must be shutdown. The 6 hours is adequate to perform a controlled shutdown while maintaining the risk of operating in this condition at an acceptable level. Once in MODE 3, the actions of condition B shall apply.

B.1

In MODES 3, 4, and 5, the probability of most accidents or transients occurring, (and their associated consequences) is significantly reduced. Therefore, there is no need to move to a different mode if one channel of RTCB is inoperable. Opening the RTCBs will not cause a transient.

Engineering judgement has determined that 48 hours is adequate to return the channel to OPERABLE status, while maintaining the risk of having the channel inoperable at an acceptable level.

B.2

In MODES 3, 4, and 5, if the inoperable channels of RTCBs cannot be restored to OPERABLE status within 48 hours, the affected RTCBs must be opened within 1 hour. This action is conservative since opening either RTCB on the other trip leg will interrupt power to the CEDMs.

The allotted Completion Time is adequate to open the affected RTCBs, while maintaining the risk of having them closed at an acceptable level.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1

Since the RTCB channels are not subject to drift, only a CHANNEL FUNCTIONAL TEST applies. Performing the CHANNEL FUNCTIONAL TEST every 92 days demonstrates proper operation of the RTCBs.

SR 3.3.8.2

Each RTCB is actuated by an undervoltage coil and a shunt trip coil. The system is designed so that either de-energizing the undervoltage coil or energizing the shunt trip coil will cause the circuit breaker to open. When a Reactor trip is required either automatically or by using the Manual Pushbuttons in the Control Room, the undervoltage coil is de-energized and the shunt trip coil is energized.

In case of manual Pushbutton, there is a rotary switch to select undervoltage coil or shunt trip coil, or both to allow testing of each coil individually.

Therefore, once per refueling, a CHANNEL FUNCTIONAL TEST is performed which tests all four sets of undervoltage coils, and all four sets of shunt trip coils, individually. This test ensures that every undervoltage coil, and every shunt trip coil is capable of performing its intended function, and that no single active failure of any RTCB component will prevent a reactor trip.

REFERENCES

1. CESSAR-DC, Chapters 7 "Instrumentation and Controls", and 15 "Accident Analysis".

2. GDC-21, Protection System Reliability and Testability.

3. GDC-23, Protection System Failure Modes.

16A.6.9 B 3.3.9 MANUAL REACTOR TRIP

B 3.3 INSTRUMENTATION

B 3.3.9 Manual Reactor Trip

BASES

BACKGROUND

Manual Reactor Trip capability is provided to permit the operator to manually trip the reactor when necessary.

Two independent sets of two adjacent pushbuttons are provided at separate locations. Each pushbutton operates one of the four RTCBs. Depressing both pushbuttons in either set will cause an interruption of power to the Control Element Drive Mechanisms (CEDMs) allowing the Control Element Assemblies (CEAs) to fall into the core.

The 2 sets of 2 criterion was selected so that no single failure in any pushbutton circuit can either cause or prevent a reactor trip.

Manual Reactor Trip pushbuttons are also provided at the reactor trip switchgear (locally) in case the Control Room pushbuttons become inoperable, or the Control Room becomes uninhabitable.

Manual Reactor Trip capability is a backup to the automatic reactor trips generated by the RPS. Without this capability, the operator would have to wait until an automatic trip setpoint was reached, selectively de-energize power supplies, or dispatch an operator to locally open the trip circuit breakers, before a reactor trip could occur. Any one of these actions could take an unnecessary amount of time.

The Manual Pushbuttons are provided with a rotary switch to select the actuation of either undervoltage coil or Shunt trip coil or both to allow testing of each coil. During normal operation the switch should be set at both position to ensure both coils actuate.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS	There are no accident analysis which take credit for the manual Reactor Trip. However, it is part of the RPS circuitry. It is to be used by the operator to shutdown the reactor whenever any parameter is rapidly trending toward its trip setpoint. Waiting and relying on the automatic system to trip the reactor is not viewed as a good operating practice.
LCO	The manual reactor trip system consists of 2 sets of 2 manual reactor trip pushbuttons located in the Control Room. Depressing either set (2 buttons) will initiate a reactor trip. The "2 sets of 2" (pushbuttons) strategy was selected to maintain the single failure criteria and to ensure that the inadvertent depressing of one button would not cause an unnecessary reactor trip (since the operator is required to depress both pushbuttons in a given set in order to cause a reactor trip).
APPLICABILITY	In MODES 1 and 2, the CEAs are at least partially withdrawn. Since this requires power to be available to the CEDMs, there is no qualifying statement placed on the APPLICABILITY about the operability of the manual reactor trip circuitry or power availability to the CEDMs. In MODES 3, 4, and 5, it is common to have the reactor trip circuit breakers open and/or power unavailable to the CEDMs. Since the CEAs are incapable of withdrawal in either condition, there is no need to have the manual reactor trip circuitry OPERABLE. Therefore, there is a qualifying statement placed on the applicability in those modes requiring the manual reactor trip circuitry be OPERABLE when the reactor trip circuit breakers are closed and power is available to the CEDMs. This ensures that the reactor remains tripable when necessary, but allows for maintenance or testing when a reactor trip is not needed. In MODE 6 the CEDMs are disconnected from their power supply and are not able to withdraw the CEAs. Therefore, manual reactor trip capability is unnecessary.

(continued)

BASES

ACTIONS

A.1

In MODES 1 and 2, if one channel becomes inoperable, the RTCBs associated with the inoperable channel must be opened within 1 hour. This action is conservative since depressing the manual reactor trip pushbutton associated with either set of breakers in the other trip leg will cause a reactor trip. With this configuration, a single channel failure will not prevent a reactor trip.

The allotted Completion Time is adequate to open the affected RTCBs, while maintaining the risk of having them closed at an acceptable level.

A.2

If the breakers associated with the inoperable channel cannot be opened, the reactor must be shutdown. The 6 hours is adequate to perform a controlled shutdown while maintaining the risk of operating in this condition at an acceptable level. Once in MODE 3, the actions of condition B shall apply.

B.1

In MODES 3, 4, and 5, the probability of most accidents or transients occurring, (and their associated consequences) is significantly reduced. Therefore, there is no need to move to a different mode if one channel is inoperable. Opening the RTCBs will not cause a transient.

The 48 hours is based on Engineering judgment and is adequate to return the channel to OPERABLE status, while maintaining the risk of having the channel inoperable at an acceptable level.

B.2

In MODES 3, 4, and 5, if the inoperable channel cannot be restored to OPERABLE status within 48 hours, the affected RTCBs must be opened within 1 hour. This action is conservative since depressing the manual reactor trip pushbutton associated with either set of breakers in the other trip leg will interrupt power to the CEDMs.

(continued)

BASES

ACTIONS (continued)	The allotted Completion Time is adequate to open the affected RTCBs, while maintaining the risk of having them closed at an acceptable level.
SURVEILLANCE REQUIREMENTS	SR 3.3.9.1 The Manual Reactor Trip circuitry is not subject to drift, therefore only a CHANNEL FUNCTIONAL TEST applies. Performing the CHANNEL FUNCTIONAL TEST prior to the reactor startup is adequate to ensure the channel is OPERABLE.
REFERENCES	1. CESSAR-DC Chapter 15 "Accident Analysis".

16A.6.10 B 3.3.10 ENGINNERED SAFETY FEATURES ACTUATION SYSTEMS (ESFAS) INSTRUMENTATION

B 3.3 INSTRUMENTATION

B 3.3.10 Engineered Safety Features Actuation Systems (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS instrumentation channels specified automatically actuate the Engineered Safety Features (ESF) system components which protect the public and plant personnel against the accidental release of fission products in the event of an accident.

The ESFAS contains devices and circuitry which generate the following signals when monitored variables reach levels that are indicative of conditions requiring protective action:

1. Safety Injection Actuation Signal (SIAS)
2. Containment Spray Actuation Signal (CSAS)
3. Containment Isolation Actuation Signal (CIAS)
4. Main Steam Isolation Signal (MSIS)
5. Loss of Voltage Signal (LOVS)
6. Emergency Feedwater Actuation Signal (EFAS)

Each of the above ESFAS actuation systems is subdivided into two separate subsystems:

1. Process Measurement Channels
2. ESFAS Logic

Process Measurement Channels for all ESFAS functions are addressed in this specification.

(continued)

BASES

BACKGROUND (continued)

ESFAS Logic, consisting of Local Coincidence Logic, Initiation Logic, and Actuation Logic, is addressed in LCO 3.3.11. Manual actuation of ESFAS is addressed in LCO 3.3.12.

A process measurement channel as defined in this specification consists of all equipment from the sensor through the Bistable Logic Processors. ESFAS process measurement channels provide the signal conditioning for the measured plant parameters used for comparison with setpoints established by transient or accident analysis. If a measured parameter exceeds its setpoint, the output from a Bistable Logic processor is forwarded to the ESFAS logic for further evaluation.

Four measurement channels with electrical and physical separation are provided for each parameter used to generate ESF actuation signals. When any two channels of like instrumentation receive a process measurement exceeding its setpoint, an ESF actuation signal is generated by the ESFAS actuation logic.

All the protection channel is isolatad from control channels through fiber optic cables. This arrangement meets the requirements of IEEE 279-1971.

Four undervoltage relays [(induction disc relays with inverse time characteristics)] are provided on each 4.16 kV Class 1E bus for detecting a loss of bus voltage. Relays are combined in a 2/4 logic to generate a LOVS. This assures a reliable power supply for each train associated with an ESF function.

APPLICABLE SAFETY ANALYSIS

The required channels of ESFAS instrumentation provide plant protection in the event of any of the analyzed accidents.

SIAS

SIAS ensures acceptable consequences during Large Break Loss of Coolant Accidents (LOCAs), Small Break LOCAs, CEA Ejection Accidents, Steam Generator Tube Ruptures, Excess Steam Demand Events, and Main Steam Line Breaks (MSLBs).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

To provide the required protection, either a High Containment Pressure or a Low Pressurizer Pressure signal will initiate SIAS.

CSAS

CSAS actuates containment spray, preventing containment over-pressurization during Large Break LOCAs, Small Break LOCAs, and MSLBs or Feedwater Line Breaks (FWLBs) inside containment.

CSAS is initiated by Hi-Hi containment pres-

CIAS

CIAS ensures acceptable consequences during Large and Small Break LOCAs, and MSLBs or FWLBs either inside or outside containment.

CIAS is initiated by low pressurizer pressure or high containment pressure.

MSIS

MSIS ensures acceptable consequences during excess steam demand events, MSLB's or FWLB's (between the steam generator and the main feedwater check valve), either inside or outside containment.

MSIS isolates both steam generators if either generator indicates a low pressure condition or a high level condition, or if a high containment pressure condition exists. This prevents an excessive rate of heat extraction and subsequent cooldown of the RCS during these events. And it also provides turbine protection in case of excessive moisture carry over.

LOVS

LOVS ensures acceptable consequences during a LOCA, MSLB, MFLB, or any other event with a coincident loss of preferred power supply.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

LOVS detects undervoltage on the 4.16 KV class 1E busses to which the diesel generator is connected. It initiates an automatic transfer to a backup and an automatic diesel generator start. Each diesel generator, when connected to its 4.16 KV bus, supplies adequate power to satisfy the minimum demands of the analyzed accidents.

EFAS

EFAS consists of two generator specific signals (EFAS 1 and EFAS 2). EFAS 1 initiates emergency feed to steam generator 1 and EFAS 2 initiates emergency feed to steam generator 2.

EFAS maintains a steam generator heat sink during a loss of MFW event, a Steam Generator Tube Rupture (SGTR) event, a MSLB or FWLB event either inside or outside containment, or any event where normal AC power or the MFW system is unavailable.

Low steam generator water level initiates emergency feed to the affected steam generator. If the affected steam generator recovers the level high enough, then the level high signal terminates the emergency feedwater flow to the affected steam generator.

LCO

The LCO ensures each of the following requirements is met:

1. An ESF function is initiated when necessary.
2. The required protection system instrumentation coincidence logic is maintained.
3. Sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance.

Allowable values specified ensure that violation of the Safety limits for the reactor core and RCS is prevented during normal operation and AOOs, and the consequences of accidents are acceptable.

(continued)

BASES

LCO (continued)

Only the Allowable Values are specified for each RPS trip function in the allowable value is specified such that the analytical limit assumed in the safety analysis is conservative including all applicable setpoint uncertainties.

Containment Pressure - High

This trip is set high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup), and not indicative of an off-normal condition. The setting is low enough to initiate the ESF functions when an off-normal condition is indicated. This allows the ESF systems to perform as expected in the Accident Analyses to mitigate the consequences of the analyzed accidents.

Containment Pressure - High - High

This trip is set high enough to allow for the Containment Cooling systems to attempt to mitigate the consequences of an accident before resorting to spraying borated water onto containment equipment. The setting is low enough to initiate CSAS to spray down the containment preventing containment pressure from exceeding design.

Pressurizer Pressure - Low

This trip is set low enough to prevent actuating the ESF functions (SIAS & CIAS) during normal plant operation and pressurizer pressure transients. The setting is high enough that with the specified accidents the ESF systems will actuate to perform as expected, mitigating the consequences of the accidents.

The Low Pressurizer Pressure trip setpoint, which provides a SIAS, CIAS, and RPS trip, may be manually decreased to a floor value, to allow for a controlled cooldown and depressurization of the RCS without causing a reactor trip, CIAS or SIAS. The margin between actual pressurizer pressure and the trip setpoint must be maintained less than or equal to the specified value to ensure a Reactor Trip, CIAS and SIAS will occur if required during RCS cooldown and depressurization.

(continued)

BASES

LCO (continued)

From this reduced setting, the trip setpoint will increase automatically as pressurizer pressure increases, tracking actual RCS pressure until the trip setpoint is reached.

When the trip setpoint has been lowered below the bypass permissive setpoint, the Low Pressurizer Pressure reactor trip, CIAS and SIAS actuation may be manually bypassed in preparation for shutdown cooling. When RCS pressure rises above the bypass permissive, the bypass is removed. This is consistent with PPS operating bypass philosophy of removing bypasses when the enabling conditions are no longer satisfied.

The bypass permissive setpoint was chosen such that events originating from below this setpoint do not require the ESF function to mitigate the event.

Steam Generator Pressure - Low

The setpoint is below the full load operating value for steam pressure so as not to interfere with normal plant operation. However, the setting is high enough to provide the required protection for a excessive steam demand event. An excessive steam demand event causes the RCS to cool down resulting in a positive reactivity addition to the core.

MSIS limits this cooldown by isolating both steam generators if the pressure in either drops below the trip setpoint. A RPS trip on Low Steam Generator Pressure is initiated simultaneously.

The Low Steam Generator Pressure trip setpoint may be manually decreased as steam generator pressure is reduced. This prevents an RPS trip or MSIS actuation during controlled plant cooldown. The margin between actual pressurizer pressure and the trip setpoint must be maintained less than or equal to the specified value to ensure a reactor trip and MSIS will occur when required.

Unlike Low Pressurizer Pressure, there is neither a floor, nor a bypass on the Low Steam Generator Pressure function.

(continued)

BASES

LCO (continued)

4.16kV Emergency Bus Undervoltage

In combination with the inverse time characteristic, the trip settings are low enough to prevent spurious trips caused by the offsite power source. Conversely, the settings are high enough to allow for detection of a loss of voltage on the 4.16kv Bus to initiate an automatic transfer to the backup preferred power source and an automatic diesel start.

APPLICABILITY

In MODES 1, 2 and 3 there is sufficient energy in the primary and secondary systems to warrant ESF system responses to:

1. Close the Main Steam Isolation valves (MSIVs) to preclude a positive reactivity addition.

2. Actuate Emergency Feedwater to preclude the loss of the steam generators as a heat sink (in the event the normal feedwater system is not available).

3. Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by not exceeding the containment design pressure, and to mitigate the effects of the accident.

4. Actuate ESF systems to ensure sufficient borated inventory to permit adequate core cooling and reactivity control during a design basis LOCA or MSLB accident.

All the following ESF functions are required to be operable in these MODES:

1. Safety Injection Actuation - SIAS

2. Containment Spray Actuation - CSAS

3. Containment Isolation - CIAS

(continued)

APPLICABILITY (continued)

4. Main Steam Line Isolation - MSIS

5. Emergency Feedwater - EFAS

6. Loss of Voltage Signal - LOVS

For MODE 4 there is sufficient energy and potential in the primary and secondary systems to warrant 1) the automatic actuation of all components to mitigate the consequences of a large break LOCA or Main Steam Line Break (MSLB) and 2) prevent or limit the release of fission product radioactivity to the environment. ESF functions which apply to Mode 4 operation follow:

1. Safety Injection Actuation - SIAS

2. Main Steam Line Isolation - MSIS

3. Containment Isolation - CIAS

4. Containment Spray Actuation - CSAS

In MODES 5, and 6 these functions are not required (except CIAS, CSAS) because adequate time is available to evaluate plant conditions and respond by manually operating the ESF components if required. In most cases, the equipment actuated by these ESFAS functions need not be operable.

ACTIONS

The Unit Review Group (URG) shall determine the desirability of maintaining any channel in bypass pursuant to Administrative Controls.

If the process variable for one ESF channel becomes inoperable, the operators are required to return the inoperable channel to service as soon as practical. The channel must be OPERABLE prior to returning to MODE 4 following entry into MODE 5. This approach to bypass/trip in four channel protection systems is consistent with the applicable criteria of IEEE standard 279.

(continued)

BASES

ACTIONS (continued)

A.1

ESFAS coincidence logic is normally 2/4. If the number of channels OPERABLE is one less than the Total Number of Channels (i.e., only 3 out of 4 available), startup or power operation is allowed to continue as long as the inoperable channel is placed in bypass or trip within 1 hour. The provision of 4 independent and redundant trip channels allows one channel to be bypassed (removed from service) during operations, placing the ESFAS in 2/3 coincidence logic. It is preferable to place an inoperable channel in bypass rather than trip, since no single additional failure can neither cause nor prevent an ESFAS actuation.

If the channel fails (or is placed) in trip, just one spurious signal from any of the other three channels will cause the ESFAS actuation. Although this is acceptable from an accident analysis standpoint, it is not good operating practice. The time allotted to bypass or trip the channel allows the operator to take all appropriate actions for the failed channel which maintaining the risk of operating with the failed channel at an acceptable level.

A.2

Some of the ESFAS instrumentation channels feed other ESFAS instrumentation channels and Reactor Protective System (RPS) instrumentation channels. If one of the ESFAS channels fails, the associated ESFAS and RPS channels should also be bypassed within 1 hour, for the same reasons, if it is determined that they are inoperable.

A.3

An inoperable channel must be returned to OPERABLE status prior to entry into an applicable MODE following the next entry into MODE 5. This is in keeping with the general philosophy of not operating the plant in an off-normal condition indefinitely. The time allowed should be adequate to repair the inoperable channel. If the channel cannot be returned to OPERABLE status by the end of the next entry into MODE 5, the problem must be taken care of before startup and power operation can commence.

(continued)

BASES

ACTIONS (continued)

B.1

With the number of channels OPERABLE one less than the Minimum Channels Operable requirement, one inoperable channel must be placed in bypass, and the other channel must be placed in trip within the required Completion Time. With one channel of ESFAS instrumentation bypassed, the ESFAS is in 2/3 logic, but with another channel failed, the ESFAS may be operating with a 2/2 logic (this assumes the channel failed in the non-conservative direction away from the trip setpoint). This is outside the assumptions made in the analyses and must be corrected. To correct the problem, the second channel is placed in trip. This places the ESFAS in a 1/2 logic. If any of the other OPERABLE channels receives a trip signal, the ESFAS function will be initiated.

B.2

Some of the ESFAS instrumentation channels feed other ESFAS instrumentation channels and Reactor Protective System (RPS) instrumentation channels. If a second ESFAS channel fails, the associated ESFAS and RPS channels should also be placed in trip or bypass within the same Completion Time as B.1, if it is determined that they are inoperable.

Operation in MODES 1 and 2 may continue until the next CHANNEL FUNCTIONAL TEST. Operation in MODES 1 and 2 cannot continue beyond that time, because performing a CHANNEL FUNCTIONAL TEST on an OPERABLE channel, will either result in an ESFAS actuation, or an unacceptable 1/1 ESFAS logic.

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

Performing a CHANNEL CHECK once every 12 hours ensures that any channel which drifts beyond the normal expected instrument drift, as compared to other identical channels, is detected within a reasonable amount of time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Data Processing System (DPS) continuously performs a cross channel comparison and will initiate an alarm to warn operators that a channel has drifted out-of-tolerance or is not working properly.

The reason the CHANNEL CHECK is performed instead of a more thorough test, is that it imposes much less of a burden on operations than any of the other instrument channel tests. If the other instrument tests are performed at their appropriate frequencies, this test is reliable.

SR 3.3.10.2

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire channel will perform its intended function when needed. It is not necessary to perform this test more often because operating experience has shown that the instrumentation stays reliable within that time frame.

Major portions of the ESFAS are monitored and/or tested by the automatic test network. Those portions of the system which are not amenable to automatic testing because they involve actuation of electromechanical devices, or involve devices which are not within the PPS cabinets, can be tested manually. The automatic test network is capable of performing tests during reactor operation. The automatic testing does not degrade the ability of the ESFAS to perform its intended function.

Detail description of the ESFAS automatic test features are described in CESSAR-DC Chapter 7.3.

SR 3.3.10.3

Performance of a CHANNEL CALIBRATION every 18 months ensures that the channels are reading accurately and within specified tolerances. Operating experience has shown this test interval to be satisfactory.

SR 3.3.10.4

Performance of a CHANNEL CALIBRATION every 18 months ensures that the operating bypasses are operating accurately and within the specified tolerances. Operating experience has shown this test interval to be satisfactory.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.10.5

Verifying the logic for the operating bypasses to be OPERABLE within 92 days prior to each reactor startup, ensures that the bypasses and their permissive setpoints function as designed prior to entering a MODE where they are required. Operating experience has shown the specified interval to be satisfactory.

REFERENCES

1. CESSAR-DC, Chapters 7 "Instrumentation and Controls", and 15 "Accident Analysis".

2. IEEE Standard 279-1971, Criteria for Protection Systems for Nuclear Power Generating Stations", April 5, 1972.

16A.6.11 B 3.3.11 ENGINNERED SAFETY FEATURES ACTUATION SYSTEMS (ESFAS) LOGIC

B 3.3 INSTRUMENTATION

B 3.3.11 Engineered Safety Features Actuation System (ESFAS) Logic

BASES

BACKGROUND

The (ESFAS) signals actuate the Engineered Safety Features (ESF) Systems to protect the public from the accidental release of fission products.

The ESFAS contains devices and circuitry which generate the following signals when monitored variables reach levels that are indicative of conditions requiring protective action.

Safety Injection Actuation	(SIAS)
Containment Spray Actuation	(CSAS)
Containment Isolation Actuation	(CIAS)
Main Steam Isolation	(MSIS)
Loss of Power	(LOVS)
Emergency Feedwater Actuation	(EFAS 1 and 2)

Each of the above ESFAS systems is subdivided into four separate subsystems:

Process Measurement Channels
Local Coincidence Logic
Initiation Logic
Actuation Logic

Process Measurement Channels for all ESFAS functions are addressed in Specification 3.3.10.

Manual actuation of ESFAS is addressed in Specification 3.3.12.

This specification addresses ESFAS Logic Coincidence, Initiation, and Actuation Logic.

ESFAS and Reactor Protection System (RPS) Local Coincidence and Initiation Logic are housed in the Plant Protection System (PPS) cabinet. ESFAS logic is very similar to RPS logic of Specification 3.3.7.

(continued)

BASES

BACKGROUND (continued)

ESFAS Actuation Logic is housed in the Emergency Safety Features - Component Control System (ESF-CCS). There are four trains for safety Injection requiring two intact trains to meet the safety analysis. Except Safety Injection, all other Emergency Safety Features are two trains thus requiring two trains of actuation signal. But in case of CSAS and EFAS, the actuation signals are arranged in four channels such that train A assigned to train A valves train B assigned to train A pumps, etc. Thus eliminating any spurious signals or failure of actuation logic channels shall not initiate the safety feature train. Both trains (four trains) are normally actuated on an automatic ESFAS initiation. Either train controls sufficient equipment to protect the public.

The following provides a brief description of ESFAS logic, as it applies to this Specification.

ESFAS Local Coincidence and Initiation Logic employ a scheme similar to the RPS, in which the four independent input channels provide an ESFAS actuation, when the bistable logic channels of more than two of the four channels sensing the same input parameter generate a trip. This is called a 2 out of 4 (2/4) trip logic. The individual parameters monitored for each ESFAS function are addressed in Specification 3.3.10.

To assure that no single failure in the RPS will either cause an inadvertent trip, or improperly prevent a trip from occurring, a minimum of 2/3 trip logic is required.

When performing maintenance, testing, or removing a failed channel from service, logic for the affected parameter(s) is changed from 2/4 to 2/3 by trip channel bypassing the affected trips. All ESFAS trips can be trip channel bypassed, providing each is bypassed in one channel at a time.

In addition to trip channel bypasses, there is also an operating bypass, shared with the RPS, on low pressurizer pressure.

(continued)

BASES

BACKGROUND (continued)

Local Coincidence logic determines if a coincidence exists in the tripping of like bistable comparators (those monitoring the same parameter) in two or more channels. Each ESFAS function will either have its own Local Coincidence logic or share Local Coincidence logic with other ESFAS functions using the same inputs.

There are four Local Coincidence Logic Channels per ESFAS Function, comparing the outputs of all four channels of bistables, taken all combinations of more than two channels Deenergizing one or more Local Coincidence Logic Channels will deenergize the associated Initiation Logic Channels.

There are four Initiation Logic channels for each ESFAS function. Each Initiation Logic Channels receiving inputs from governing Local Coincidence Logic channels. The manual actuation signals (described in LCO 3.3.12) are OR gated here with signals generated by Local coincidence Logic channels. The initiation Logic channels send outputs into two or four channels of Actuation Logic channels (ESF-CCS) depending on the number of actuation logic channels specified in the Table 3.3.11-1.

Each of the two channels (or four channels) of Actuation Logic for each ESFAS function consists of a selective 2/4 logic, actuating component control logic for subgroup outputs for the ESF equipment.

This logic is implemented by interconnecting the four ESFAS outputs into two parallel trip legs with two initiation outputs in each trip leg. Tripping either output in both trip legs will actuate the ESFAS train. The actuation signals from Actuation Logic Channels actuates the component control logics in the ESF-CCS which actually govern the valves and pumps.

Since a single failure neither causes nor prevents the protection system actuation, and protection channels are isolated from control channels through fiber optic cables, this arrangement meets the requirements of IEEE 279.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS

The ESFAS logic provides for automatic ESF initiation. All transients and accidents that call for ESF actuation assume the ESFAS logic is functioning as designed.

The bases for the individual ESFAS functions are stated in specification B 3.3.10.

LCO

The LCO on the ESFAS logic channels ensures that all of the following requirements are met:

1. The required protection system coincidence logic is maintained (minimum 2/3 logic, normal 2/4 logic).

2. Sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance.

APPLICABILITY

The ESFAS Local Coincidence Logic Channels must be OPERABLE whenever automatic actuation of the ESFAS function is required.

The initiation and actuation logic channels must be OPERABLE when either a manual or automatic actuation is required. Manual actuation is effected by opening contacts in the initiation logic channels. Each of the four manual pushbuttons actuates one initiation logic channel. Therefore, initiation logic and downstream actuation logic must be OPERABLE.

This ensures the ESFAS function can be actuated when required, but allows for maintenance or testing when an ESFAS actuation is not required.

ACTIONS

Normal ESFAS logic is 2/4. In case of trip channel bypass any input parameter in accordance with Specification 3.3.10, the logic of Local Coincidence Logic Channels are rendered into 2/3.

This approach to bypass/trip in a four-channel protection system is consistent with the applicable criteria of IEEE Standards 279-1971.

(continued)

BASES

ACTIONS (continued)

Failures of individual Bistable Logic Channels are addressed in LCO 3.3.10. This specification addresses failures of the Local Coincidence, Initiation and Actuation Logic not addressed in the above.

A.1

With one Local Coincidence Logic Channels except for SIAS and MSIS channels inoperable, the channels must be restored to OPERABLE status within 48 hours.

This allows the operator time to take appropriate actions while ensuring any risk involved in operating with a failed channel is acceptable.

A.2.1 and A.2.2

If action A.1 cannot be completed within the required Completion Time, the reactor must be brought to a MODE where the action statement does not apply. The 12 hours to be in MODE 4 is adequate to perform a controlled shutdown of the plant while maintaining the risk of operating in this condition at an acceptable level.

B.1

With one Local Coincidence Logic channels for SIAS or MSIS inoperable, the channels must be restored to OPERABLE status within 48 hours, for the same reasons as A.1.

C.1

With one Initiation Logic channel inoperable, the channel must be restored to OPERABLE status within 48 hours.

This provides the operator time to take appropriate actions while ensuring that any risk involved in operating with a failed channel is acceptable.

(continued)

BASES

ACTIONS (continued)

D.1

With two initiation logic channels in the same trip leg inoperable, it is necessary to open contacts in that trip leg in trains of the affected ESFAS function(s). This must be done in a timely manner to assure the ESF function can be actuated if required. This is done by opening at least one of the two initiating relay contacts in the affected trip leg. If these contacts have failed open, such as during the failure of a Power Supply, the action is satisfied.

The 1 hour allowed to open at least one contact in the affected trip leg of ESFAS Actuation Logic trains, is adequate to perform the action, while maintaining the risk of having the contacts closed at an acceptable level.

D.2

The inoperable channels must be restored to OPERABLE status within 48 hours. The allowed time is adequate to take appropriate actions while ensuring that any risk involved in operating with the failed channel(s) is acceptable.

E.1 and E.2

If the Required Action of Condition B, C, or D, cannot be completed within the required Completion Time, the reactor must be brought to a MODE where the action statement does not apply.

The 36 hours to be in MODE 5 is adequate to perform a controlled shutdown and cooldown of the plant while maintaining the risk of operating in this condition, at an acceptable level.

(continued)

BASES

ACTIONS (continued)

F.1 and F.2

NOTE

Each actuation logic channel is responsible for initiating one train of the ESF function(s). Surveillance testing may temporarily render an individual train inoperable. This is acceptable providing the other train is OPERABLE. 1 hour is adequate to complete required testing, while ensuring any risk involved is acceptable.

F.1 and F.2

With one train of ESFAS inoperable, a single failure in the other train could render the entire function inoperable. For this reason, the reactor must be safely brought to a MODE where adequate time exists to manually actuate any ESF components which may be required.

The 12 hours to be in MODE 4 is adequate to perform a controlled shutdown of the plant while maintaining the risk of being in this condition, at an acceptable level.

SURVEILLANCE REQUIREMENTS

SR 3.3.11.1

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire channel will perform its intended function. Operating experience has demonstrated that the instrumentation remains reliable within this time frame.

Major portions of the ESFAS are monitored and/or tested by the automatic test network. Those portions of the system which are not amenable to automatic testing because they involve actuation of electromechanical devices, or involve devices which are not within the PPS cabinets, can be tested manually. The automatic test network is capable of performing tests during reactor operation. The automatic testing does not degrade the ability of the ESFAS to perform its intended function.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

Detail descr... of the ESFAS automatic test features are described in CESSAR-DC Chapter 7.3.

SR 3.3.11.2

Subgroup outputs are responsible for actuating the specific ESF equipment when they are deenergized by the selective 2/4 actuation logic. Channel Functional Testing (SR 3.2.11.1) tests the ESFAS through the initiation signals from Initiation Logic in the Actuation Logic, but does not test individual subgroup outputs.

This surveillance requires the testing of all subgroup outputs from selective 2/4 actuation logic which can be tested during power operation every 182 days. Those outputs which cannot be tested during plant operation must be tested during MODE 5.

Engineering judgment has determined test interval to be adequate to ensure operability of these relays.

SR 3.3.11.3

The logic for Trip Channel Bypasses must be verified operable within 92 days prior to each reactor startup. The inability to Trip Channel Bypass could adversely affect ESFAS reliability, since without this capability the ESFAS may be placed in 1/3 logic during channel testing or maintenance. This increases the likelihood of an inadvertent ESFAS actuation.

Failure of a Trip Channel Bypass contact to open upon bypass removal will defeat the Local Coincidence Logic maintaining 2/3 logic rather than 2/4 logic in the affected parameter. Depending upon the specific failure, it might then be possible to trip channel bypass another channel in the same parameter, placing the ESFAS in 2/2 logic, which is unacceptable.

REFERENCES

1. CESSAR-DC, Chapters 7 "Instrumentation and Controls", and 15 "Accident Analysis".

2. IEEE Standard 279-1971, Criteria for Protection Systems for Nuclear Power Generating Stations", April 5, 1972.

16A.6.12 B 3.3.12 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) INSTRUMENTATION - MANUAL ACTUATION

B 3.3 INSTRUMENTATION

B 3.3.12 Engineered Safety Features Actuation System (ESFAS)Instrumentation - Manual Actuation

BASES

BACKGROUND	Manual ESFAS initiation capability is provided to permit the operator to manually actuate an Engineered Safety Features (ESF) system when necessary. Two sets of two pushbuttons (located in the Control Room) for each ESF function are provided, and each set actuates both trains. In this specification, each set of adjacent pushbuttons is designated as a single channel. The manual ESFAS capabilities are a backup to the automatic ESF actuations generated by the ESFAS. Without this capability, the operator would have to wait until either an automatic initiation trip setpoint was reached, or the equipment required was selectively energized before an ESF response could occur. Either of those actions could take a considerable amount of time.
APPLICABLE SAFETY ANALYSIS	There is no accident analysis which specifically takes credit for manual ESFAS actuation. However, it is part of the ESFAS circuitry, is housed in the Plant Protective System (PPS) cabinet, and is used by the operator to actuate an associated ESF function whenever any parameter is rapidly trending toward its trip setpoint. For these reasons it is included in the Technical Specifications.
LCO	The LCO ensures the proper amount of reliability and redundancy is maintained in the manual ESFAS actuation circuitry to ensure the operator has manual ESFAS initiation capability. The "2 sets of 2" pushbutton strategy was selected to maintain the single failure criteria (IEEE Std 279-1971), and to ensure that inadvertent depressing of one button would not cause an unnecessary ESFAS initiation.

(continued)

APPLICABILITY

CSAS

In MODES 1, 2, 3 and 4, there is sufficient energy in the primary and secondary systems to warrant the actuation of Containment Spray to mitigate the consequences of a Large Break LOCA, Main Steam Line Break, or Main Feedwater Line Break inside containment. Therefore, the specified channels of manual actuation must be OPERABLE in these MODES.

EFAS

In MODES 1, 2 and 3, Emergency Feedwater is required for those events where normal feedwater is not available, to preclude loss of steam generators as a heat sink. In MODE 4, EFAS may not be required because shutdown cooling can be used to remove decay heat.

SIAS, CIAS, CSAS and MSIS

In MODES 1, 2, 3 and 4 there is sufficient energy in the primary and secondary systems to warrant the actuation of SIAS, CIAS, and CSAS to mitigate the consequences of a Large Break Loss Of Coolant Accident (LBOCA), or high energy line break (steam or feedwater) inside or outside containment. Therefore, the specified channels of manual actuation must be OPERABLE in these MODES.

ACTIONS

A.1

If the number of channels OPERABLE is one less than the Total Number of Channels of the table, the inoperable channel must be restored to OPERABLE status within 48 hours, or the plant must be brought to a MODE where the LCO does not apply.

The time allowed to restore the channel to OPERABLE status is based upon engineering judgment. It is adequate to perform the Required Action, while keeping the risk of operating with the inoperable channel at an acceptable level, since there is one channel left to manually actuate the ESF function.

(continued)

BASES

ACTIONS (continued)

A.2.1 and A.2.2

The time allowed to bring the plant to MODE 4, is based upon engineering judgment and is adequate to shutdown the plant in a controlled manner, while maintaining the risk of operating with the inoperable channel at an acceptable level.

SURVEILLANCE REQUIREMENTS

SR 3.3.12.1

A CHANNEL FUNCTIONAL TEST is performed every 18 months to ensure the channels will perform their intended function when needed. Since the channels are not subject to instrument drift and there are no meters for indication, no other tests need to be performed. The interval is based upon engineering judgment and operating experience, and is adequate to ensure channel OPERABILITY.

REFERENCES

1. CESSAR-DC, Chapters 7 "Instrumentation and Controls", and 15 "Accident Analysis".
2. IEEE Standard 279-1971, Criteria for Protection Systems for Nuclear Power Generating Stations", April 5, 1972.

16A.6.13 B 3.3.13 REMOTE SHUTDOWN MONITORING INSTRUMENTATION (RSMI)

B 3.3 INSTRUMENTATION

B 3.3.13 Remote Shutdown Monitoring Instrumentation (RSMI)

BASES

BACKGROUND

The specified remote shutdown monitoring instrumentation provides the necessary information at the Remote Shutdown Panel (RSP) to allow the operator to monitor the status of the plant while bringing it to MODE 4 or MODE 5, in the event that the Control Room becomes uninhabitable.

In the unlikely events that the control room is uninhabitable sufficient instrumentation and controls are provided (per 10CFR50 Appendix A, Criteria 11 (Ref. 3) to achieve prompt hot standby of the reactor, maintain the unit in a safe condition during hot shutdown, and achieve cold shutdown using the RSP and other local control stations.

Before the evacuation of the control room, the reactor should be tripped and the control capability for equipments listed in the Table 3.3.13-1 shall be transferred to RSP disabling the control room station. Thus removing any faulty signals interfering from control room.

At least one measurement channel (per specified parameter), similar to the four measurement channels provided for control room operation and protection, provides information at the RSP. The measurement channels are provided with electrical and physical separation for each parameter displayed.

APPLICABLE SAFETY ANALYSIS

There are no transient or accident analyses which take credit for operations conducted from the Remote Shutdown Panel.

The required RSP and associated instrumentation is consistent with General Design Criterion 19 of 10CFR50.

LCO

The LCO on RSP instrumentation ensures that adequate information is available to allow the operator to monitor and control the status of the plant (outside Control Room) while bringing it to MODE 4 or MODE 5.

(continued)

BASES

APPLICABILITY

The specified instrumentation is the minimum required to allow the operator to monitor the status of the plant while bringing it to MODE 4 or MODE 5 from MODES 1, 2, or 3. Therefore, it must be OPERABLE in MODES 1, 2, and 3.

ACTIONS

A.1

If the number of channels OPERABLE becomes less than the Minimum Channels Operable requirement of table 3.2.13-1, the inoperable channel(s) must be restored to OPERABLE status within 7 days, or the plant must be brought to a MODE where the LCO does not apply.

The allowed time to restore the instrument channel to OPERABLE status is based upon engineering judgment, and is adequate to perform the Required Action while keeping the risk of being without the channel at an acceptable level. Considered in the assignment of this allowed time is the fact that other instrumentation channels (not directly reading the parameter that the failed channel had been reading) can give approximate indication or trends of the parameter that the failed channel had been reading.

B.1

The 12 hours to be in MODE 4 if action A.1 cannot be met, is adequate to shutdown the plant from MODE 1, while maintaining the risk of operating with the failed channel at an acceptable level.

SURVEILLANCE REQUIREMENTS

SR 3.3.13.1

Performing a CHANNEL CHECK once every 31 days ensures that any channel which drifts beyond the normal expected instrument drift, as compared to other identical channels, is detected within a reasonable amount of time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

The reason the CHANNEL CHECK is performed instead of a more thorough test, is that it imposes much less of a burden on operations than any of the other instrument channel tests. If the other instrument tests are performed at their appropriate frequencies, this test is reliable.

SR 3.3.13.2

Performance of a CHANNEL CALIBRATION every 18 months ensures that the channels are reading accurately and within specified tolerances. Operating experience has shown this test interval to be satisfactory.

REFERENCES

1. CESSAR-DC, Chapters 7 "Instrumentation and Controls", and 15 "Accident Analysis".

2. IEEE Standard 279-1971, Criteria for Protection Systems for Nuclear Power Generating Stations", April 5, 1972.

3. General Design Criterion 19 of 10CFR50.

16A.6.14 B 3.3.14 ACCIDENT MONITORING INSTRUMENTATION (AMI)

B 3.3 INSTRUMENTATION

B 3.3.14 Accident Monitoring Instrumentation (AMI)

BASES

BACKGROUND

The Accident Monitoring Instrumentation provides the necessary information to assist the operator in monitoring post-accident conditions within the RCS, the steam generating system, and the containment.

The Regulatory Guide 1.97 classified the accident monitoring instrumentation into five categories (type A, B, C, D, E) based on the purpose of each monitored variables (eg. to permit the operator to take manual actions - type A, indications whether safety functions are being accomplished ~~0~~-type B, etc.) and the variables are classified into category 1, 2, 3 according to their importance (1 for Key parameters, 2 for Status of safety systems, 3 for status of support systems). The category 1 variables are required to be monitored in two channels.

The seismically qualified Discrete Indication and Alarm System (DIAS) channel P is dedicated to continuously monitor and display the category 1 parameters. The DIAS channel N and Data Processing System (DPS) also monitor all the category 1, 2, and 3 parameters as a backup for channel P.

Two measurement channels provide the necessary information in the Control Room for adequate accident monitoring. The channels provide wide-range information which meet electrical and physical separation requirements for each parameter displayed. This design is consistent with the requirements of IEEE 279-1971. The channels are provided with equipment qualified to operate in the environments specified for design basis events in the CESSAR-DC. These channels comply with the requirements of NUREG 0578 and recommendations of Regulatory Guide 1.97.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS

All transients and accidents evaluated assume monitoring and surveillance capabilities exist for specific parameters to assess the event in progress, and the Reactor Protective System (RPS) and Engineered Safety Features (ESF) systems responses to the event. The required Accident Monitoring Instrumentation supplies parameter information that is reliable and adequate to provide the operator with this capability during normal and post-accident conditions.

LCO

The LCO ensures that adequate and reliable information is available to allow the operator to monitor the progress of any event and the performance of systems required to mitigate the effects of the event.

The channel requirements listed in Table 3.3.14-1 provide the minimum required instrumentation (category 1 per Reg. Guide 1.97) necessary for adequate and reliable post-accident plant monitoring. These requirements are consistent with the recommendations of Regulatory Guide 1.97, and are derived from the Combustion Engineering CEN 152 Emergency Operating Instruction (EOI) Guidelines.

APPLICABILITY

The specified instrumentation is the minimum required to allow the operator to monitor the post accident status of the plant and ESF systems while bringing it to MODE 4 or MODE 5 from MODES 1, 2, or 3. Therefore it must be OPERABLE in MODES 1, 2, and 3 and as specified in TABLE 3.3.14-1.

ACTIONS

A.1

If the number of channels OPERABLE becomes less than the Total Number of Channels requirement, but greater than or equal to the Minimum Channels Operable requirement of the table, the inoperable channel(s) must be restored to OPERABLE status within 7 days, or the plant must be brought to a MODE where the LCO does not apply.

The allowed time to restore the proper redundancy of Accident Monitoring Instrumentation is adequate, while maintaining the risk of having only the Minimum Channels Operable requirement met, at an acceptable level.

(continued)

BASES

ACTIONS (continued)

B.1

If the Minimum Channels Operable requirement of the table cannot be met, the affected channel(s) must be restored to OPERABLE status within 48 hours, or the plant must be brought to a MODE where the LCO does not apply. The remaining instrumentation, while not giving exact readings of the parameter which was being monitored by the failed channel, still gives indication of the trend of the parameter. Therefore, immediate plant shutdown is not warranted. However, if the Minimum Channels Operable requirement cannot be restored within 48 hours, the risk of operating in this condition would require the plant to be shut down.

The time allowed to restore the Minimum Channels Operable requirement is adequate, while maintaining the risk of not having the Minimum Channels Operable requirement met, at an acceptable level.

C.1

The 12 hours allotted to bring the plant to MODE, 4 if the Required Actions are not met, is adequate to allow for a controlled plant shutdown from MODE 1, while maintaining the risk of operating without the Minimum Channels Requirement met, at an acceptable level.

SURVEILLANCE REQUIREMENTS

SR 3.3.14.1

Performing a CHANNEL CHECK once every 31 days ensures that any channel which drifts beyond the normal expected instrument drift, as compared to other identical channels, is detected within a reasonable amount of time.

The reason the CHANNEL CHECK is performed instead of a more thorough test, is that it imposes much less of a burden on operations than any of the other instrument channel tests. If the other instrument tests are performed at their appropriate frequencies, this test is reliable.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.14.2

Performance of a CHANNEL CALIBRATION every 18 months ensures that the channels are reading accurately and within specified tolerances. Operating experience has shown this test interval to be satisfactory.

REFERENCES

1. CESSAR-DC, Chapters 7 "Instrumentation and Controls", and 15 "Accident Analysis".

2. IEEE Standard 279-1971, Criteria for Protection Systems for Nuclear Power Generating Stations", April 5, 1972.

3. General Design Criterion 19 of 10CFR50.

4. CEN 152 Rev.3 "Combustion Engineering Emergency Operating Instruction Guidelines".

5. CESSAR-DC, Section 15.8 "Shutdown Risk Report"

16A.7 B 3.4 REACTOR COOLANT SYSTEM

16A.7.1 B 3.4.1 RCS PRESSURE, TEMPERATURE AND FLOW DNBR LIMITS

B 3.4 REACTOR COOLANT SYSTEM

B 3.4.1 RCS Pressure, Temperature and Flow DNBR Limits

BASES

BACKGROUND

This bases addresses requirements for maintaining Reactor Coolant System (RCS) pressure, loop temperature, and loop flow rate within limits assumed in the safety analysis so that the acceptance criteria for the Departure from Nucleate Boiling Ratio (DNBR) will be met in the event of transients.

The accident analyses of normal operating conditions and Anticipated Operational Occurrences (AOOs) assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters assure that these parameters will not be less conservative than was assumed in the analyses and thereby provide assurance that the minimum DNBR will meet required criteria for each of the transients analyzed.

The LCO for the minimum and maximum RCS pressure as measured at the pressurizer are consistent with operation within the nominal operating envelope and are bounded by those used as the initial pressures in the analyses.

The LCO for minimum and maximum RCS cold leg temperature are consistent with operation at the indicated power level and are bounded by those used as the initial temperatures in the analyses.

The LCO for minimum and maximum RCS flow are bounded by those used as the initial flow rates in the analyses. The RCS flow rate is not expected to vary during plant operation with all pumps running.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the plant safety analysis (Ref. 1). The safety analysis has shown that transients initiated from the limits of LCO 3.4.1 will meet the DNBR criteria. These transients include loss of coolant flow events and dropped or stuck control rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.7, regulating rod insertion limits, LCO 3.2.~~[illegible]~~ 3, azimuthal power tilt, and LCO 3.2.5, axial shape index.

(continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)	LCO 3.4.1, pressure, temperature, and flow DNB limits, satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 2), because they limit the variation of RCS pressure, temperature and flow, which are initial condition inputs to the plant safety analysis.
LCOs	LCO 3.4.1 provides limits on the monitored process variables pressurizer pressure, RCS cold leg temperature, and RCS total flow rate to ensure that the core operates within the limits assumed for the plant safety analysis. Operating within these limits will result in meeting DNBR criteria in the event of a DNB limited transient.
APPLICABILITY	In MODE 1 and 2, the limits on RCS pressure, RCS cold leg temperature, and RCS flow rate must be maintained in order to assure that DNBR criteria will be met in the event of an unplanned loss of coolant flow or other DNBR limiting transient. In all other MODES, DNBR is not a concern because the power level is low, providing other applicable LCOs are met.
ACTIONS	A.1 Pressurizer pressure is a controllable and measurable parameter. With this parameter not within the LCO limits, action must be taken to restore the parameter. The two-hour completion time is based on plant operating experience that shows the parameter can be restored in this time period. RCS flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the flow rate is not within the LCO limit, then power must be reduced, as required in ACTION B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds. The two-hour completion time for restoration of the parameter provides sufficient time to determine if the violation of an LCO limit is due to instrument error.

(continued)

BASES

ACTIONS (continued)

B.1

If Required Action A.1 is not met within the associated completion time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 in six hours. The six hours is a reasonable time that permits the plant power to be reduced at an orderly rate in conjunction with even control of steam generator heat removal. In MODE 3, the reduced power condition reduces the potential for violation of the accident analysis bounds.

C.1

RCS cold leg temperature is a controllable and measurable parameter. With this parameter not within the LCO limits, action must be taken to restore the parameter. The two-hour completion time is based on plant operating experience that shows that the parameter can be restored in this time period.

D.1

If Required Action C.1 is not met within the associated completion time, the plant is placed in a mode where the LCO does not apply. This is done by placing the plant in MODE 3 within six hours. The six-hour completion time is a reasonable time that permits power reduction at an orderly rate in conjunction with even control of steam generator heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

The 12-hour surveillance of pressurizer pressure is sufficient to ensure that the pressure can be restored to a normal operation, steady-state condition following load changes and other expected transient operations. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analysis assumptions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.2

The 12-hour surveillance of RCS cold leg temperature is sufficient to ensure that the RCS coolant temperature can be restored to a normal operation, steady-state condition following load changes and other expected transient operations. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analysis assumptions.

SR 3.4.1.3

The 12-hour surveillance of RCS total flow rate is performed using the installed flow instrumentation. This surveillance verifies RCS flow within the bounds of the analyses. The 12-hour interval has been shown by operating experience to be sufficient to assess degradation and verify generation within safety analysis assumptions.

This surveillance is modified by a note which only requires performance of this SR in MODE 1. The note is necessary to allow measurement of RCS flow at normal operating conditions at power.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a [precision calorimetric heat balance or pump ΔP method] once every 18 months allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow is within the bounds of the analyses.

The intent of the surveillance frequency of 18 months is to reflect the importance of reverifying flow after a refueling outage where the core has been altered which may have caused an alteration of flow resistance.

The surveillance is modified by a note which states that SR 3.0.4 is not applicable. The note is necessary to allow measurement of the flow rate at normal operating conditions at power in MODE 1.

(continued)

BASES

REFERENCES 1. CESSAR-DC Chapter 15, Accident Analysis.

2. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

16A.7.2 B 3.4.2 RCS MINIMUM TEMPERATURE FOR CRITICALITY

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

Establishing the value for the minimum temperature for reactor criticality is based upon considerations for: 1) operation within the existing instrumentation ranges and accuracies, and 2) operation within the bounds of the existing accident analyses. The reactor protection system receives inputs from the narrow range hot leg temperature detectors which have a range of [520°F to 620°F], and the integrated control system controls average temperature (T_{avg}) using inputs of the same range. Nominal temperature T_{avg} for making the reactor critical is [543°F]. Theoretically there is no specific minimum temperature design constraint for making the reactor critical. There do not appear to be any fundamental material or equipment limitations which would prevent adoption of a lower minimum. However, selection of instrument ranges and analysis inputs was done in anticipation of [543°F] being the minimum temperature at which criticality would occur. Safety and operating analyses for lower temperatures have not been made. Plants have not been licensed for low temperature criticality and licensing regulations permitting criticality below the normal power operating range have not been developed for commercial power reactors.

APPLICABLE SAFETY ANALYSES

There are no accident analyses which dictate the minimum temperature for criticality, but safety analyses assumed initial temperatures no lower than [543°F].

This specification preserves limits used in the safety analysis and therefore satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 1).

LCOs

The purpose of the LCO is to prevent criticality outside the normal operating regime [543°F - 565°F]. While it is theoretically possible to operate the reactor at critical conditions at lower temperatures, specific design features have been included and analyses have been performed on the basis that it is neither necessary nor desirable to do so. Consequently, this LCO prevents operation in an unanalyzed regime.

(continued)

BASES

LCOs (continued)	The LCO is only applicable below [550°F] and provides a reasonable distance to the limit of [543°F]. This allows adequate time to trend its approach and take corrective actions prior to exceeding the limit.
APPLICABILITY	The reactor has been designed and analyzed to be critical in MODES 1 and 2 only, and in accordance with this specification, criticality is not permitted in any other MODE. Therefore, this LCO is applicable in MODE 1 and MODE 2 when $K_{eff} \geq 1.0$. Coupled with the applicability definition for criticality is a temperature limit. Monitoring is required at and below a T_{avg} of [550°F]. The no-load temperature of [557°F] is maintained by the steam dump control system.
ACTIONS	A.1 With T_{avg} below 543°F, restoration is required within 15 minutes. The Completion Time of 15 minutes restricts the period for operation outside the analyzed limits. The Completion Time is reasonable for the operator to accomplish the specified actions. A.2 If the Required Action is not met within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in MODE 3 in 30 minutes. The allowed time reflects the urgency of maintaining the plant within the analyzed range.
SURVEILLANCE REQUIREMENTS	SR 3.4.2.1 T_{avg} is required to be verified above [543°F] within 15 minutes prior to achieving criticality and every 30 minutes thereafter when the MODE requirements apply. The 15-minute time period allows the operator to adjust temperatures or delay criticality so the LCO will not be violated. The 30-minute time is frequent enough to prevent inadvertent violation of the LCO.

(continued)

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BASES

SURVEILLANCE REQUIREMENTS (continued)	While surveillance is required whenever the reactor is critical and temperature is at or below [550°F], in practice the surveillance is most appropriate during the period when the reactor is brought critical. Because the operator would likely verify average RCS temperature more often than required by this surveillance, it is less restrictive than normal operating practice.
REFERENCES 1.	52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

16A.7.3 B 3.4.3 RCS PRESSURE AND TEMPERATURE (PT) LIMITS

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

Pressure and Temperature (P/T) limit curves for heatup, cooldown, and Inservice Leak and Hydrostatic testing (ISLH), and data for the maximum allowable rate of change of reactor coolant temperature are in the Pressure and Temperature Limits Report (PTLR). The heatup curve provides both heatup limits and criticality limits.

The PT limit curves define an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup and cooldown maneuvering where loop temperature and pressure indications are monitored and compared to the curves to determine that operation is within the allowable region. The limit for the allowable rate-of-change of temperature is similarly monitored by predicting the temperature change over a fixed time period and comparing it to the limit.

The purpose of this LCO is to establish operating limits that provide a wide margin to non-ductile (brittle) failure of major piping and pressure vessel components of the Reactor Coolant Pressure Boundary (RCPB). Of the major components within the RCPB, the reactor vessel, outlet nozzles, and head are the components most subject to brittle failure and therefore are the components for which the technical specification limits are most pertinent.

The origin of the P/T limits is found in Appendix G to 10 CFR 50 (Ref. 1). Appendix G requires that limits be established and the limits shall be based on specific fracture toughness requirements for RCPB materials such that an adequate margin to brittle failure will be provided during operational occurrences. 10 CFR 50 Appendix G mandates the use of ASME III, Appendix G (Ref. 2).

The concern addressed by 10 CFR 50 Appendix G is that undetected flaws could exist in the RCPB components which, if subjected to unusual pressure and/or thermal stresses, could result in non-ductile failure. Certain RCS P/T combinations can create stress concentrations at flaw locations. If the stress concentrations are of sufficient magnitude, flaw growth can result in failure before the ultimate strength of the material is attained. Flaw growth is

(continued)

BASES

BACKGROUND (continued)

resisted by the material toughness and toughness can cause flaw growth to be arrested. Toughness is a property that varies with temperature and is lower at room temperature than operating temperature. Furthermore, the material toughness is affected by neutron fluence which causes the steel ductility to decrease. The effect of fluence is cumulative and ductility steadily decreases with exposure time. Only the vessel beltline region is in a high fluence area. Toughness is also dependent on the chemistry of the base metal, weld metal, and heat affected zone metal and their impurities.

One indicator used to indicate the temperature effect on ductility is the Nil-Ductility Temperature, NDT (formerly called the Nil-Ductility Transition Temperature, NDTT). The NDT for the steel alloy used in vessel fabrication has been established by testing. The NDT is a temperature below which non-ductile (brittle) fracture failure may occur. Ductile failure may occur above the NDT. The exact temperature value cannot be determined very precisely. Consequently a reference temperature (RT_{NDT}) has been established by experimental means. The neutron embrittlement effect on the material toughness is reflected by increasing the RT_{NDT} as exposure to neutron fluence increases. In effect, the temperature at which brittle failure can occur increases. Regulatory Guide 1.99 (Ref. 3) provides guidance for evaluating the effect of neutron fluence. To assist in evaluating the amount of RT_{NDT} shift to be applied, surveillance specimens, made up of samples of reactor vessel material, are periodically withdrawn and analyzed.

As the RT_{NDT} increases with vessel exposure to fluence and the material toughness decreases, the P/T limit curves are correspondingly adjusted, thus giving limits that provide pressure boundary protection over the design life of the vessel. The effect of the RT_{NDT} shift is to cause the pressu- to decrease at a given temperature.

This specification provides two types of limits:

- Reactor coolant P/T curves that define allowable operating regions.
- Limits on the allowable rate-of-change of temperature of the reactor coolant which provide limits on the thermal gradients through the walls of the vessel and thus limits tensile stresses in the vessel wall.

(continued)

BASES

BACKGROUND (continued)

In use, the P/T curves are primarily for prevention of non-ductile failure, whereas the rate-of-change of temperature limits assist in prevention of both ductile and non-ductile failure.

The three curves (heatup, cooldown, and ISLH) are composite curves established by superimposing limits derived from stress analyses for those portions of the reactor vessel and head that are most restrictive. At any specific pressure, temperature, and temperature rate-of-change, one location within the geometry of the reactor vessel or head will dictate the most restrictive limit. Across the entire pressure and temperature span of the limit curves, different locations are most restrictive and thus the curves are composites of the most restrictive regions.

The heatup curves represent a different set of restrictive elements than the cooldown curves because the thermal gradients through the vessel wall are reversed. The thermal gradient reversal tends to alter the location of the tensile stress from outer to inner walls. The ISLH curve values use different calculation safety factors (per ASME Appendix G) from the heatup and cooldown curves.

The ISLH curves also extend to the higher pressure (~2500 psig) to bound the test range. The curves have been developed for heatup, ISLH testing, and cooldown in conjunction with stress analyses to allow a large number of operating cycles and also provide a conservative margin to non-ductile failure. The heatup limit curve also contains a limit defining the minimum PT for criticality (note also that LCO 3.4.2 specifies a more restrictive minimum temperature for criticality than these limits which are based on 10 CFR 50, Appendix G).

This specification requires a post-event evaluation if the limits are violated. The evaluation may take different forms depending on the severity of the violation and can include: comparisons to existing pre-analyzed transients already contained in the stress analysis, new stress analysis, component inspection, or other. One method that may be used is the guidance given by ASME XI Appendix E (Ref. 4). Appendix E is simplified and permits a quick review, but it is limited in application (only the vessel beltline). Although the P/T limits have been created primarily for monitoring the vessel and head, a severe violation may indicate a need to also review the condition of other RCS components.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The limits are not derived from design basis accidents presented in the CESSAR DC ~~FSAR~~, but are prescribed as guidance used during normal operation to avoid encountering pressure, temperature, and temperature rate-of-change conditions which might cause undetected flaws to propagate, resulting in non-ductile failure of the RCPB.

Except as noted below

(Insert G)

Linear Elastic Fracture Mechanics (LEFM) methodology, following the guidance given by 10 CFR 50 Appendix G, ASME III Appendix G, and Regulatory Guide 1.99, is used to determine the stresses and material toughness at locations within the RCPB. Although any region within the pressure boundary is subject to non-ductile failure, the regions that provide the most restrictive limits are the vessel closure head, the outlet nozzles, and the vessel beltline. With increasing neutron fluence, the vessel beltline becomes the most restrictive region.

A number of analytical steps comprise the overall analyses that establish the limits. The following summarizes the basic elements:

1. Define the temperature profile for heatup and cooldown. The reactor coolant temperature rate-of-change is defined so that normal plant operation can readily proceed without constraint. Cooldown and ISLH rates-of-change have been similarly defined. These rates-of-change become LCO limits as well as the basis for heat transfer calculations.

2. Perform heat transfer calculations to determine the thermal gradient through the walls. The analyses account for variance of flow rate and consequent changes in the rate of heat transfer between the reactor coolant and the walls during different stages of heatup and cooldown when the number of operating reactor coolant pumps change.

3. Establish the material toughness as a function of RT_{NDT}. ASME Section III, Appendix G provides the basis for RT_{NDT} and Regulatory Guide 1.99 provides the basis for adjusting RT_{NDT} as a function of neutron fluence and materials constituents and impurities.

(continued)

INSERT G

Steam line break and other increased heat removal events require a SIAS on low pressurizer pressure to ensure subcriticality via boration for events postulated to be initiated at relatively high RCS temperatures. The pressurizer temperature will not drop sufficiently to cause a SIAS for these events if the combination of pressurizer pressure and temperature is not maintained above the limit specified by the region of unallowed operation in Figures 3.4.3-1A and B.

This should go into the BASES for 3.4.3

BASES

APPLICABLE SAFETY ANALYSES (continued)

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating the reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied to the adjacent section of the reactor vessel. The limit curves must be recalculated when the RT_{NDT} determined from the surveillance capsule is different from the calculated RT_{NDT} for the equivalent capsule radiation exposure.

4. Perform a LEFM analysis to establish the pressure and temperature limits. Stress analyses are performed and the criteria for setting the limits is that the combined temperature and pressure stresses cannot exceed the material toughness for the specific temperature under examination. Analytical stress concentration at each location under examination is driven by postulating specific flaw sizes. Stress intensity factors for pressure and temperature are calculated and are compared to a reference stress intensity factor. Safety factors are applied to the pressure stress intensity factor.

5. Measurement Adjustment - The curves are adjusted for differences in elevation between the instrumentation tap location and the location of interest (beltline, etc.) and are adjusted for the system pressure losses for the number of reactor coolant pumps that are operated at different stages of heatup or cooldown.

   Instrument errors are estimated and the curves include adjustments to pressure and temperature.

In Reference 6, the NRC specified that the RCS P/T limits specification met Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 5) and the LCO should be retained in Technical Specifications.

BASES

LCOs

The two elements of the LCO are:

1. The limit curves for a) heatup, b) cooldown, and c) ISLH, and

2. Limits on the rate-of-change of temperature.

The rate-of-change of temperature limits control the thermal gradient through the walls and is used as input for calculating the heatup, cooldown and ISLH limit curves. Thus, the LCO for the rate-of-change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violation of the limits places the reactor vessel outside of the bounds of the stress analysis and can increase stresses in reactor coolant system components. The consequences to the reactor vessel and other RCS components depends on several factors including the severity of the departure from the allowable operating pressure temperature regime or the severity of the rate of change of temperature. The consequences also depend on the length of time that the limits were violated (longer violations allow the temperature gradient in the thick walls of the vessel to become more pronounced), and the consequences also depend on the existence, size and orientation of flaws in the vessel material. Although vessel failure is not an expected outcome of a violation, the possibility for failure exists.

APPLICABILITY

The NRC staff believes the concern for non-ductile (brittle) failure exists at all times. The RCS P/T limits specification provides a definition of acceptable operation for prevention of non-ductile failure that is in accordance with 10 CFR 50 Appendix G (Ref. 1). Although the P/T limits were developed to provide guidance for operation during heatup, cooldown (MODES 3, 4, and 5) and ISLH testing, their applicability is to be at all times in keeping with the concern for non-ductile failure. At all times is defined to be any condition with fuel in the reactor vessel.

However, during MODES 1 and 2, other LCOs provide limits for operation that can be more restrictive than the P/T limits. These other LCOs include LCO 3.4.2, RCS minimum temperature for criticality, and LCO 3.4.1, RCS pressure temperature and DNB limits. SL 2.1, safety limits for pressure and

(continued)

APPLICABILITY (continued)

temperature and maximum pressure also provide operational restrictions. In MODE 6, with the reactor vessel head detensioned or removed, the capability for violating the P/T curves does not exist, however the potential for violating the temperature rate-of-change limit remains.

Furthermore, in MODES 1 and 2, operation is above the temperature range of concern for non-ductile failure. As such, stress analyses have been developed in accordance with normal maneuvering profiles such as power ascension.

ACTIONS

The actions designated by this specification are based on the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, which may be accompanied by equipment failures, may also require additional actions based on emergency operating procedures.

A.1 and A.2

With operation not within the limits of the LCO, restoration within the limits is required because the RCPB must be placed into a condition that has been verified by stress analysis. The required action is in the proper direction to reduce RCPB stress.

The completion time of 30 minutes reflects the urgency of restoring the parameter(s) to within the analyzed range. Most violations will not be severe and the activity can be accomplished in this time in a controlled manner. However, if the activity cannot be accomplished, then the subsequent Required Actions B.1 and B.2 require further pressure and temperature reduction.

In addition to restoration an evaluation to determine if RCS operation may proceed is required. The purpose of the evaluation is to determine if RCPB integrity remains acceptable and must be accomplished prior to continuing operation. A variety of methods may be used for the evaluation including a comparison to pre-analyzed transients accounted for in the stress analysis, new analyses, or inspection of the components. ASME Appendix E may be used to support the evaluation, however, its use is restricted to evaluation

(continued)

The NOTE to Action A.2 eliminates the requirement for this evaluation when the operation extends to the "Region of Unallowed Operation" since the RCS P-T limits have not been exceeded.

BASES

ACTIONS (continued)

of the vessel beltline. If the evaluation cannot be accomplished in 72 hours, or if the results of the evaluation are indeterminate or unfavorable, then the next appropriate action is to proceed to further reduce pressure and temperature as given in Required Actions B.1 and B.2.

The 72-hour completion time is a reasonable time to accomplish the necessary activities. For a mild violation, the evaluation should be possible within this time. As part of the evaluation it may be desirable to determine what an appropriate rate of cooldown might be or if a soak period is desirable. More severe violations may require special, event specific stress analyses and/or inspections which are appropriately carried out while the RCS is in a reduced pressure and temperature condition as specified by Required Actions B.1 and B.2.

The Note which applies to Conditions A and B requires that all required actions must be completed whenever either or both conditions are entered. The purpose of the note is to give additional emphasis to the need to restore operation to the allowable condition and to perform an evaluation of the effects of any excursion outside of the allowable limits. Restoration alone is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If the Required Action is not met within the associated Completion Time, the plant must be placed in a lower operating MODE. Reducing the MODE is considered a prudent action because: a) the RCS remained in an unacceptable region for an extended period of increased stress, or b) a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, which is best accomplished while the RCS is in a low pressure and temperature state. With the plant at reduced pressure conditions, the possibility of propagation of undetected flaws is reduced.

The six-hour time for achieving MODE 3 is a reasonable time to reach MODE 3 from full power without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

The 36-hour completion time for achieving MODE 5 is reason- able based on operating experience to reach the required MODF from full power without challenging plant systems. The time permits an orderly cooldown and a soak period, if needed, or a slower average rate of cooldown (~5°F/hr). A soak period may be desirable if the temperature rate of change limit has been violated. The pressure limit of 500 psig corresponds to the Low Temperature Overpressurization Limit (LTOP).

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within limits is required when RCS temperature and pressure conditions are undergoing planned changes. The time period of 30 minutes is based on industry-accepted practice. Since temperature rate-of-change limits are specified in hourly increments, a half hour time period permits assessment and correction for minor deviations within a reasonable time. Surveillance for heatup and cooldown, and ISLH may be discontinued when definitions given in the plant procedures for defining the end of these conditions are satisfied.

The surveillance is modified by a note which states that the surveillance is only required during heatup, cooldown, and ILSH testing. There are no surveillance requirements during critical operation because LCO 3.4.2, RCS minimum temperature for criticality, contains a more restrictive LCO.

REFERENCES

1. 10 CFR 50, Appendix G, "Fracture Toughness Requirements."

2. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure."

3. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May, 1988.

4. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section XI, Appendix E, "Evaluation of Unanticipated Operating Events."

(continued)

BASES

REFERENCES (continued)

5. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

6. Letter from T. E. Murley (USNRC) to J. K. Gasper (C-E Owners' Group) transmitting "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Group's Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," dated May 9, 1988.

7. CESSAR-DC, Section 19.8 "Shutdown Risk Report"

16A.7.4 B 3.4.4 RCS LOOPS - MODES 1 AND 2

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODES 1 and 2

BASES

BACKGROUND

The Reactor Coolant System (RCS) uses two reactor coolant pumps (RCPs) per steam generator loop and two steam generator loops. The pump flow rate has been sized to provide core heat removal with appropriate margin to departure from nucleate boiling (DNB) during power operation and for anticipated transients originating from power operation. This specification requires two RCS loops with both RCPs in each loop. The intent of the specification is to require core heat removal with forced flow during power operation. Specifying two loops provides the minimum necessary paths (two steam generators) for heat removal.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the design bases accident initial conditions including: RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate which is represented by the number of RCS loops in service.

Both transient and steady state analyses have been performed to establish the effect of flow on DNB. The transient accident analysis for the plant has been performed assuming four RCPs are in operation. The majority of the plant safety analysis is based on initial conditions at high core power or zero power. The accident analyses which involve RCP misoperation are the four pump coastdown, single pump locked rotor, and single pump broken shaft events.

Steady state DNB analysis has been performed for the four pump combination. For four pump operation, the steady state DNB analysis, which generates the DNBR limit assumes a maximum power level of [113%] RATED THERMAL POWER (RTP). This is the design overpower condition for four pump operation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The number of loops and the associated RCS flow as represented by the number of pumps in operation satisfies the requirements of Selection Criterion 2 of the Interim Policy Statement (Ref. 1), because the flow is an initial condition for transient and steady state analyses.
LCOs	The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by having both RCS loops with both RCPs in each loop in operation for removal of heat by the two steam generator loops. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power. Operation in these MODES implies that important components are OPERABLE, and an OPERABLE loop consists of RCPs providing forced flow for heat transport and steam generators which are OPERABLE in accordance with the steam generator tube surveillance program. Steam generator, and hence RCS loop, OPERABILITY with regard to SG water level is ensured by the Reactor Protection System (RPS) in MODES 1 and 2. A reactor trip places the plant in MODE 3 if any SG level is ≤ [44.2%]WR as sensed by the RPS. The minimum water level to declare the SG OPERABLE is [44.2%]WR. Operation in other MODES is covered by LCOs 3.4.5 (MODE 3), 3.4.6 (MODE 4), 3.4.7 and 3.4.8 (MODE 5), and 3.9.4 and 3.9.5 (MODE 6 - Refueling).
APPLICABILITY	The LCO is applicable in MODES 1 and 2 because forced flow for core heat removal with acceptable margin to DNB must be maintained within the acceptance criteria of the safety analysis. To ensure the safety analysis assumptions remain valid, the specification only permits operation in MODES 1 and 2 with both RCS loops and all four RCPs in operation. In MODES 3, 4, 5, and 6, DNB is not limiting when the reactor is shutdown, hence the LCO is not applicable.

BASES

ACTIONS

A.1

If the required number of loops is not in operation, the Required Action is to reduce power and bring the plant to MODE 3. The action lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits. It should be noted that the reactor will trip and place the plant in MODE 3 as soon as the reactor protection system senses less than four RCPs operating.

The six hours allowed is a reasonable time based on operating experience to reach MODE 3 from full power without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This surveillance requires verification of the required number of loops in operation and reactor coolant circulation every 12 hours to ensure that forced flow is providing heat removal. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analysis assumptions. The verification may be performed by checking RCPs in operation and RCS flow and temperature indications.

SR 3.4.4.2

This SR provides the means necessary to determine steam generator OPERABILITY in an operational MODE. The requirement to demonstrate steam generator tube integrity in accordance with the Steam Generator Inspection Program emphasizes the importance of steam generator tube integrity. Even though this surveillance can not be performed at normal operating conditions, its inclusion in this specification provides a method of determining steam generator OPERABILITY during normal operating conditions.

REFERENCES

1. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

16A.7.5 B 3.4.5 RCS LOOPS - MODE 3

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

The primary function of the reactor coolant system loops in MODE 3 is removal of decay heat and transfer of this heat, via the steam generators, to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 3, reactor coolant pumps (RCPs) are used to provide forced circulation heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to satisfy single failure criteria. Only one RCP need be OPERABLE to declare the associated RCS loop OPERABLE.

Reactor coolant natural circulation is not normally used, however, the natural circulation flow rate is sufficient for core cooling. However, natural circulation does not provide a rapid response in the RCS to changes in conditions. Boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be assured.

APPLICABLE SAFETY ANALYSES

Only the inadvertent deboration and startup of a RCP events safety analyses are performed with initial conditions in MODE 3. Operation of one RCP was credited on the analysis of the inadvertent deboration event. For the inadvertent startup of an RCP, not more than two RCPs were assumed to be in operation. (If two RCPs were running, they were assumed to be in the same loop.)

Failure to provide heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path which functions or actuates to prevent or mitigate a design basis accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such this LCO satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 1).

(continued)

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BASES

LCOs

The purpose of this LCO is to require two RCS loops to be available for heat removal thus providing redundancy. The LCO requires the two loops to be OPERABLE with the intent of requiring both steam generators to be capable (≥ [25%]WR water level) of transferring heat from the reactor coolant at a controlled rate. Forced reactor coolant flow is the required way to transport heat, although natural circulation flow provides adequate removal. A minimum of one running RCP meets the LCO requirement for one loop in operation.

The LCO note permits a limited period of operation without RCPs. This means that natural circulation has been established. When in natural circulation, boron reduction is prohibited because an even concentration distribution throughout the RCS cannot be assured. Core outlet temperature is to be maintained at least 10°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

In MODES 3, 4, and 5, it is sometimes necessary to stop all RCPs or shutdown cooling (SDC) pump forced circulation (e.g., change operation from one SDC train to the other, perform surveillance or startup testing, perform the transition to and from SDC system cooling, or to avoid operation below the RCP minimum NPSH limit). The time period is acceptable because natural circulation is adequate for heat removal, or the reactor coolant temperature can be maintained subcooled and boron stratification affecting reactivity control is not expected.

Operation in this MODE implies that components are OPERABLE, and an OPERABLE loop consists of a RCP providing forced flow for heat transport and a steam generator which is OPERABLE in accordance with the steam generator tube surveillance program and has the minimum water level for SG OPERABILITY.

APPLICABILITY

In MODE 3, the heat load is lower than at power and one RCP is adequate for transport. Two loops are required for redundancy for heat removal.

Operation in other MODES is covered by LCO 3.4.4 (MODES 1 and 2), 3.4.6 (MODE 4), 3.4.7 and 3.4.8 (MODE 5), and 3.9.4 and 3.9.5 (MODE 6 - Refueling).

(continued)

BASES

ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for heat removal may have been lost.

The required action is restoration of the RCS loop to OPERABLE status within a completion time of 72 hours. This time allowance is based on engineering judgment considering that a single loop has a heat transfer capability much greater than needed to remove the decay heat produced in the reactor core.

B.1

If restoration is not possible within 72 hours, the unit must be placed in MODE 4. In MODE 4 the plant may be placed on the shutdown cooling System. The allowed Completion Time of 12 hours is reasonable based on operating experience to reach the required MODE from the existing plant condition without challenging plant systems.

C.1 and C.2

If no loop is in operation, except as provided in Note 1 in the LCO section, all operations involving a reduction of RCS boron concentration must be immediately suspended. This is necessary because boron dilution requires forced circulation for proper homogenization. Action to restore one RCS loop to operation shall be immediately initiated and continued until one RCS loop is restored to operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This surveillance requires verification of the required RCS loop in operation and reactor coolant circulation every 12 hours to ensure forced flow is providing heat removal. Verification includes flow rate and temperature monitoring. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analysis assumptions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.2

This surveillance requires verification of water level in each steam generator $\geq$ [~~44.2~~] 25 %WR every 12 hours. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within the safety analysis assumptions.

SR 3.4.5.3

Verification that the required number of reactor coolant pumps are OPERABLE ensures that additional reactor coolant loops can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs. The Frequency of seven days is an accepted industry practice and has been shown to be acceptable by operating experience.

REFERENCES

1. 52 FR 3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States nuclear Regulatory Commission, February 6, 1987.

16A.7.6 B 3.4.~~7~~ 6 RCS LOOPS - MODE 4

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the Reactor Coolant System (RCS) loops is the removal of decay heat and transfer of this heat to the steam generator(s) or shutdown cooling system (~~SDC~~ SCS) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 4, either reactor coolant pumps (RCPs) or SDC divisions can be used for coolant circulation. The intent of this LCO is to provide forced flow from at least one RCP or one ~~SDC~~ SCS division for decay heat removal and transport. The flow provided by one RCP or ~~SDC~~ SCS division is adequate for heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for heat removal.

This LCO permits limited periods without forced circulation. When RCPs are stopped, the steam generator heat removal provides a natural circulation flow rate that is sufficient for decay heat removal

When the ~~SDC~~ SCS pumps are stopped, no alternate heat removal path exists, unless the RCS and steam generators have been placed in service in forced or natural circulation. The response of the RCS without the ~~SDC~~ SCS system depends on the core decay heat load and the length of time that the ~~SDC~~ SCS pumps are stopped. As decay heat diminishes, the effects on RCS temperature and pressure diminish. Without cooling by ~~SDC~~ SCS, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (pressure and temperature limits or low temperature overpressurization limit) must be observed and forced ~~SDC~~ SCS flow or heat removal via the steam generators must be reestablished prior to reaching the pressure limit.

(continued)

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BASES

BACKGROUND (continued)

Entry into a condition with no ~~SDC~~ SCS divisions in operation should only be considered for limited circumstances which include: 1) a heat removal path(s) via the RCS and steam generator(s) is in operation, or 2) pressure and temperature increases are easily maintained within the allowable pressure and subcooling limits.

APPLICABLE SAFETY ANALYSES

The only safety analyses performed with initial conditions in MODE 4 are the inadvertent deboration and inadvertent startup of RCP events. No forced coolant circulation was credited for the inadvertent deboration event. For the inadvertent startup of an RCP, not more than two RCPs were assumed to be in operation. (If two RCPs were running, they were assumed to be in the same loop.)

Failure to provide heat removal may result in challenges to a fission product barrier. The RCS loops or ~~SDC~~ SCS divisions are a part of the primary success path which functions or actuates to prevent or mitigate a design basis accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, the LCO satisfies the requirements of Criterion 3 of the NRC interim Policy Statement (Ref. 1).

LCOs

The purpose of this LCO is to require the availability of a minimum of two RCS loops/~~SDC~~ SCS divisions for heat removal thus providing redundancy. The LCO allows the two loops/divisions that are required to be OPERABLE to be comprised of any combination of RCS loops or ~~SDC~~ SCS divisions.

The LCO note permits a limited period of operation without RCPs. This means that natural circulation has been established using the steam generators. With the RCS in natural circulation, boron reduction is prohibited because an even concentration distribution throughout the RCS cannot be assured. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

The LCO Note also permits the ~~SDC~~ SCS divisions to be stopped. The circumstances for stopping both ~~SDC~~ SCS divisions are to be limited to: 1) situations where pressure and pressure and temperature increases can be maintained well within the allowable pressure (PT and LTOP) and 10°F

(continued)

LCOs (continued)

subcooling limits, or 2) an alternate heat removal path(s) through the steam generator(s) is in operation. The LCO Note prohibits boron dilution when SCS ~~SDC~~ forced flow is stopped because an even concentration distribution cannot be assured.

[350°F] during cooldown or ~~and~~ [290°F] during heatup (the heatup rate is limited to [40°F/hr or less])

The second LCO Note requires that ~~either of~~ the following ~~two~~ condition ~~s~~ be satisfied before an RCP may be started with any RCS cold leg temperature ≤ [317°F]:

secondary water temperature in each SG must be < [100°F] above each of the RCS cold leg temperatures.

Satisfying ~~either of the above~~ this condition ~~s~~ will preclude ~~a large pressure surge in the RCS~~ violating RCS P/T Limits. (see LCO 3.4.11) when the RCP is started.

In MODES 3, 4, and 5, it is sometimes necessary to stop all RCP or SCS ~~SDC~~ pump forced circulation (i.e. change operation from one SCS ~~SDC~~ division to the other, perform surveillance or startup testing, perform the transition to and from SCS ~~SDC~~, or to avoid operation below the RCP minimum NPSH limit). The time period is acceptable because natural circulation is adequate for heat removal or the reactor coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.

Operation in this MODE implies that components are OPERABLE, and an OPERABLE RCS loop consists of an RCP providing forced flow for heat transport and a steam generator which is OPERABLE in accordance with the steam generator tube surveillance program and has the minimum water level for SG OPERABILITY. Similarly, for the SCS ~~SDC~~ division, the SCS ~~SDC~~ pump(s) are capable of providing forced flow for heat exchange.

APPLICABILITY

In MODE 4, this LCO applies because it is possible to remove core decay heat with either the RCS loops and steam generators or the ~~SDC System.~~ SCS.

Operation in other MODES is covered by LCOs 3.4.4 (MODES 1 and 2), 3.4.5 (MODE 3), 3.4.7 and ~~3.9.4 and 3.9.5~~ 3.4.8 (MODE 5), ~~and 3.8.5~~ 3.9.4 and 3.9.5 (MODE 6 - Refueling).

(continued)

BASES

ACTIONS

A.1

If only one required RCS loop is OPERABLE, redundancy for heat removal is lost. The required action is to initiate activities to restore a second loop/division to OPERABLE status and the action must be taken ~~within 15 minutes.~~ immediately. Even though one loop/division is OPERABLE and in operation, the completion time emphasizes the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If only one required ~~SDC~~ SCS division is operable, redundancy for heat removal is lost. The required action is to restore a second loop/division to OPERABLE status within one hour. Even through one division is OPERABLE and in operation, the completion time emphasizes the importance of maintaining the availability of two paths for heat removal. If a second loop/division cannot be restored to an OPERABLE status within one hour, the plant must be placed in MODE 5 within the next 24 hours. Placing the plant in MODE 5 is a conservative action with regard to decay heat removal. With only one ~~SDC~~ SCS division OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining SDC division, it would be better to initiate that loss from MODE 5 (≤ ~~200°F~~ 210°F) rather than MODE 4 (~~200~~ 210 - ~~300~~ 350°F). The completion time of 25 hours is reasonable based on operating experience to reach MODE 5 from MODE 4, with only one SDC division operating, without challenging plant systems.

C.1 and C.2

If no RCS loops or ~~SDC~~ SCS divisions are in operation, the action requires immediate suspension of any operation for boron reduction and requires action to immediately start restoration of one operating loop/division. The action for restoration does not apply to the condition for no loops in operation when the exemption Note in the LCO is in force. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop/division is restored to operation.

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This surveillance requires verification of water level in the required steam generator(s) ≥ 25% every 12 hours. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analysis assumptions.

SR 3.4.6.2

This surveillance requires verification of the required loop/division in operation every 12 hours to ensure forced flow is providing heat removal. Verification of RCS or ~~SDC~~ SCS operation includes flow rate and temperature monitoring. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analysis assumptions.

SR 3.4.6.3

Verification that the required number of pumps are OPERABLE ensures that additional pumps can be placed in operation, if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pumps. The Frequency of seven days is accepted industry practice and has been shown to be acceptable by operating experience.

REFERENCES

1. 52 FR 3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," USNRC, 2/6/87.

2. Generic Letter 88-17, "Loss of Decay Heat Removal," USNRC, 10/17/88.

3. CESSAR-DC, Section 19.8 "Shutdown Risk Report"

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16A.7.7 B 3.4.7 RCS LOOPS - MODE 5 (LOOPS FILLED)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5 (Loops Filled)

BASES

BACKGROUND

In MODE 5 with the Reactor Coolant System (RCS) loops filled, the primary function of the RCS loops is the removal of decay heat and transfer of this heat to the steam generator(s) or shutdown cooling (~~SDC~~ SCS) heat exchangers. While the principle means for decay heat removal is via SDC, the steam generators are specified as a backup means for redundancy. Even through the steam generators cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the steam generator water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the ~~SDC~~ SCS divisions are the principle means for heat removal. The number of divisions in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one ~~SDC~~ SCS division or RCS loop for decay heat removal and transport. The flow provided by one RCP or ~~SDC~~ SCS division is adequate for decay heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RCS loop or ~~a SDC~~ SCS division which must be OPERABLE and in operation. The second path can be another OPERABLE RCS loop or SDC division, or maintaining an adequate water level in each steam generator.

This LCO permits limited periods without forced circulation. When the ~~SDC~~ SCS pumps are stopped, no alternate heat removal path exists unless the RCS and steam generators have been placed in service in forced or natural circulation. The response of the RCS without the ~~SDC system~~ SCS depends on the core decay heat load and the length of time that the ~~SDC~~ SCS divisions are stopped. As decay heat diminishes the effects on RCS temperature and pressure diminish. Without cooling by SDC, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat

(continued)

BASES

BACKGROUND (continued)

load. Because pressure can increase, applicable system pressure limits (pressure and temperature limits or low temperature overpressurization limit) must be observed and forced ~~SDC~~ SCS flow must be reestablished prior to reaching the pressure limit. Entry into a condition with no ~~SDC~~ SCS divisions in operation should only be considered for limited circumstances which include: 1) heat removal path(s) via the RCS and steam generator(s) is in operation, or 2) pressure and temperature increases are easily maintained within the allowable pressure and subcooling limits.

APPLICABLE SAFETY ANALYSES

The only safety analyses performed with initial conditions in MODE 5 are the inadvertent deboration and inadvertent startup of an RCP events. No forced coolant circulation was credited in the inadvertent boration event. For the inadvertent startup of an RCP, not more than two RCPs were assumed to be in operation. (If two RCPs were running, they were assumed to be in the same loop.)

Failure to provide heat removal may challenge the integrity of a fission product barrier. The ~~SDC System~~ SCS or RCS loops are a part of a primary success path which functions or actuates to prevent or mitigate a design basis accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, the LCO satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 1).

LCOs

The purpose of this LCO is to require the availability of a minimum of two paths for heat removal thus providing redundancy. The LCO allows the two paths that are required to be OPERABLE to be comprised of combinations of ~~SDC~~ SCS divisions and/or the RCS loops and associated steam generators.

The LCO Note 1 permits all ~~SDC~~ SCS pumps and RCPs to be stopped. The circumstances for stopping both ~~SDC~~ SCS divisions are to be limited to: 1) situation where pressure and temperature increases can be maintained well within the allowable pressure (PT and LTOP) and [10°F] subcooling limits, or 2) an alternate heat removal path(s) through the steam generator(s) is in operation. The LCO Note prohibits boron dilution when ~~SDC~~ SCS forced flow is stopped because an even concentration distribution cannot be assured.

(continued)

BASES

LCOs (continued)

Core outlet temperature is to be maintained at least [10°F] below saturation temperature so that no vapor bubble would form and possibly cause a natural circulation flow obstruction. In this MODE, the steam generators can be used as a backup for ~~SDC~~ SCS heat removal. To ensure their availability, the RCS loop flow path is to be maintained with subcooled liquid to ensure their availability.

In MODES 3, 4, and 5, it is sometimes necessary to stop all RCP or ~~SDC~~ SCS forced circulation (i.e., change operation from one ~~SDC~~ SCS division to the other, perform surveillance or startup testing, perform the transition to and from the SCS ~~SDC~~, or to avoid operation below the RCP minimum NPSH limit). The time period is acceptable because natural circulation is acceptable for heat removal or the reactor coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.

The second LCO Note requires that ~~either of~~ the following ~~two~~ condition~~s~~ be satisfied before an RCP may be started with any RCS cold leg temperature ≤ ~~[312°F]~~:

[259°F] during cooldown or [290°F] during heatup (the heatup rate is limited to [40°F/hr or less])

a. secondary water temperature in each SG must be < [100°F] above each of the RCS cold leg temperatures.

Satisfying ~~either of the above~~ this condition~~s~~ will preclude ~~a large pressure surge in the RCS~~ violating RCS P/T Limits (see LCO 3.4.11) when the RCP is started.

The third LCO Note permits an orderly transition from Mode 5 to Mode 4 during a planned heatup by permitting removal of SCS divisions from operation when at least one RCP is in operation.

Operation in this MODE implies that components are OPERABLE, and an OPERABLE RCS loop consists of a steam generator that can perform as a heat sink (i.e., has an adequate water level), and is OPERABLE in accordance with the steam generator tube surveillance program. RCPs are OPERABLE if they are capable of being powered and are able to provide flow if required. The ~~SDC System~~ SCS is OPERABLE when it is capable of providing forced flow for heat exchange.

(continued)

BASES

APPLICABILITY

In MODE 5 with loops filled, this LCO applies because it is possible to remove decay heat with the ~~SDC System~~ SCS, but the steam generators may be used as an alternate heat sink.

Operation in other MODES is covered by LCOs 3.4.4 (MODES 1 and 2), 3.4.5 (MODE 3), 3.4.6 (MODE 4), 3.4.8 (MODE 5 - Loops Partially Filled), and 3.9.4 and 3.9.5 (MODE 6 - Refueling).

ACTIONS

A.1 and A.2

If only one required means of decay heat removal is OPERABLE, redundancy for heat removal is lost. The Required Action is to initiate activities to restore a second loop/division to OPERABLE status and the action must be taken ~~within 15 minutes~~ immediately. An alternative to restoring a second loop/division would be to initiate actions to restore the water level in the required steam generators and the action must be taken ~~within 15 minutes~~ immediately. Either Required Action A.1 or A.2 will restore redundant decay heat removal paths. Even though one loop/division is OPERABLE and in operation, the completion time emphasizes the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If both required loops/divisions are inoperable or not in operation, the action requires immediate suspension of any operation for boron reduction and requires action to immediately start restoration of one OPERABLE loop/division. The action for restoration does not apply to the condition for no loops in operation when the exemption Note in the LCO is in force. The immediate completion time reflects the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop/division is restored.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

To ensure that the steam generators are available as a backup to the ~~SDC system~~ SCS, steam generator water level is verified every 12 hours, ~~The note requires the surveillance~~ when the LCO requirement is being met by use of the steam generators. If both ~~SDC trains~~ divisions are OPERABLE, the surveillance is not needed. The 12-hour interval ~~has been~~ shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analysis assumptions.

SR 3.4.7.2

This surveillance requires verification of the required number of loops/division in operation every 12 hours to ensure forced flow is providing heat removal. Verification of operation includes flow rate and temperature monitoring. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analysis assumptions.

SR 3.4.7.3

This surveillance requires

Verification that the required number of pumps are OPERABLE ensures that additional pumps can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pumps. The Frequency of seven days is accepted industry practice and has been shown to be acceptable by operating experience.

REFERENCES

1. 52 FR 3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," USNRC, 2/6/87.

2. Generic Letter 88-17, "Loss of Decay Heat Removal," USNRC, 10/17/88.

3. CESSAR-DC, Section 19.8 "Shutdown Risk Report"

BASES

APPLICABLE SAFETY ANALYSES

The only safety analyses performed with initial conditions in MODE 5 with loops not filled are the inadvertent deboration and inadvertent startup of an RCP events. For this analysis one ~~SDC~~ SCS division was credited as operating. The flow provided by one ~~SDC~~ SCS division is adequate for heat removal and for boron mixing.

Failure to provide heat removal may result in challenges to a fission product barrier. The ~~SDC system~~ SCS is part of the primary success path which functions or actuates to prevent or mitigate a design basis accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, this LCO satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 1).

LCO

The purpose of this LCO is to require the availability of a minimum of two ~~SDC~~ SCS divisions for heat removal thus providing redundancy. An OPERABLE division is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat removal cannot occur via the ~~SDC system~~ SCS unless forced flow is used. A minimum of one running ~~SDC~~ SCS pump meets the LCO requirement for one division in operation.

LCO Note 1 allows an exception to the LCO to permit surveillance testing.

The LCO Note 2 permits the ~~SDC~~ SCS pumps to be stopped for up to 15 minutes. It is seldom necessary to stop both pumps, however, it may be necessary to stop both pumps for a short period when switching from one division to the other. The circumstances for stopping both ~~SDC~~ SCS pumps are to be limited to a short period of time and when pressure and temperature increases can be maintained within the allowable pressure (LTOP or PT) and subcooling limits.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the ~~SDC system.~~ SCS.

Operation in other MODES is covered by LCOs 3.4.4 (MODES 1 and 2), 3.4.5 (MODE 3), 3.4.6 (MODE 4), 3.4.7, (MODE 5 - Loops Filled), ~~and~~ 3.9.4 and 3.9.5 (MODE 6 - Refueling), and 3.10.4 (MODES 5 & 6 - Reduced RCS Inventory - Heat Removal)

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BASES

ACTIONS

A.1

If only one required ~~SDC~~ SCS division is OPERABLE, redundancy for heat removal is lost. The action is to initiate activities to restore a second division to OPERABLE status and the action must be taken ~~within 15 minutes.~~ immediately. Even though one loop is OPERABLE and in operation, the completion time emphasizes the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If both required ~~SDC~~ SCS divisions are inoperable or the required division is not in operation, the action requires immediate suspension of any operation for boron reduction and requires action to immediately start restoration of one division to OPERABLE status. The action for restoration does not apply to the condition of divisions not in operation when the exemption NOTE in the LCO is in force. The immediate completion time reflects the importance of maintaining operation for decay heat removal. The action to restore must be continued until one division is restored.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This surveillance requires verification of the required division in operation every 12 hours to ensure forced flow is providing heat removal. Verification of ~~SDC~~ SCS operation is performed by flow rate and temperature monitoring. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

SR 3.4.8.2

Verification that the required number of pumps are OPERABLE ensures that additional pumps can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pumps. The frequency of seven days is accepted industry practice and has been shown to be acceptable by operating experience.

(continued)

BASES

REFERENCES

1. 52 FR 3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," USNRC, 2/6/87.

2. Generic Letter 88-17, "Loss of Decay Heat Removal," USNRC, 10/17/88.

3. CESSAR-DC, Section 15.8 "Shutdown Risk Report"

16A.7.9 B 3.4.9 PRESSURIZER

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The maximum water level limit has been established to ensure that a liquid-to-vapor interface exists to permit Reactor Coolant System (RCS) pressure control, using the sprays and heaters, during normal operation and proper pressure response for anticipated design basis transients. The water level limit serves two purposes:

Pressure control during normal operation maintains subcooled reactor coolant in the loops and thus, in the preferred state for heat transport, and

By restricting the level to a maximum, expected transient reactor coolant volume increases (pressurizer insurge) will not cause excessive level changes which could result in degraded ability for pressure control.

The maximum level limit permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, thus, both sprays and heaters can operate to maintain the design operating pressure. The level limit also prevents filling the pressurizer (water solid) for anticipated design basis transients, thus assuring that pressure relief devices (code safety valves) can control pressure by steam relief rather than water relief. If the level limits were exceeded prior to a transient that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the design safety limit of 2750 psia or damage may occur to the pressurizer code safety valves.

The requirement to have two groups of pressurizer heaters assures that RCS pressure can be maintained. The pressurizer heaters maintain RCS pressure to maintain the reactor coolant subcooled. Inability to control RCS pressure during natural circulation flow could result in a loss of single phase flow and a decreased capability to remove core decay heat.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. No safety analyses are performed in lower MODES with the exception of the inadvertent deboration and inadvertent startup of an RCP events. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of non-condensible gases normally present. The steam bubble limits the volume of non-condensible gases.

Safety analyses presented in the CESSAR-DC do not take credit for pressurizer heater operation, however, an implicit initial condition assumption of the safety analyses is that the pressurizer is operating in the range of [1905 to 2375 psia].

The maximum level limit is of prime interest for the Feedwater line break event with loss of offsite power (FLBLOP). Conservative safety analyses assumptions for this event indicate that it produces the largest increase in pressurizer level. Thus, this event has been selected to establish the pressurizer water level limit. Assuming proper response action by emergency systems, the level limit prevents water relief through the pressurizer safety valves. Since prevention of water relief is a goal for abnormal transient operation rather than a safety limit, the value for pressurizer level is nominal and is not adjusted for instrument error.

The requirement for emergency power supplies is based on NUREG-0737 (Ref. 2). The intent is to allow maintaining the reactor coolant in a subcooled condition with natural circulation at hot, high pressure conditions for an undefined, but extended, time period after a loss of offsite power. While loss of offsite power is an initial condition or coincident event assumed in many accident analyses, maintaining hot, high pressure conditions over an extended time period is not evaluated as part of ~~FSAR~~ CESSAR-DC accident analyses.

The maximum pressurizer water level limit satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 1) because it prevents exceeding the initial reactor coolant mass which is an input assumption of the safety analysis. The maximum water level also permits the pressurizer code safety valves to relieve steam for anticipated pressure increase transients, preserving their function for mitigation. Thus, Selection Criterion 3 is also indirectly applicable.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated by the NRC in NUREG-0737 (Ref. 2), is the reason for providing an LCO. The heaters do not meet any of the Selection Criteria of the NRC Interim Policy Statement (Ref. 1). However, in Reference 3, the NRC maintained that the pressurizer met Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 1) and that the LCO (including pressurizer heaters) be retained in Technical Specifications.

LCOs

The purpose of the LCO is to ensure pressurizer OPERABILITY for pressure control for normal power operation and for anticipated design basis events as previously described. The intent of the LCO is to ensure that a steam bubble exists in the pressurizer to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions.

The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The exact design value of [200 kW] is derived from the use of [4] heaters rated at [50 kW] each. The needed amount to maintain pressure is dependent on the losses. Tests indicate that pressurized heat losses do not usually impose a need for [200 kW].

The need for RCS pressure control is most pertinent when core heat can cause the greatest effect on reactor coolant system temperature resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1, 2, and 3. In MODES 1, 2, and 3, the need to maintain the availability of pressurizer heaters and their emergency power supplies is most pertinent. In the event of a loss of offsite power, the initial conditions of these MODES gives the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODES 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the decay heat removal system is inservice and therefore the LCO is not applicable.

(continued)

BASES

ACTIONS

A.1 and A.2

With pressurizer water level outside the limit, action must be taken to restore the plant to operation within the bounds of the safety analysis. This is done by placing the plant in MODE 3 with the reactor trip breakers open within six hours, ~~ad~~ and placing the plant in MODE 4 within an additional six hours. This takes the plant out of the applicable MODES and restores the plant to operation within the bounds of the safety analysis.

Six hours is a reasonable time based on operating experience to reach MODE 3 from full power without challenging plant systems and operators. Further pressure and temperature reduction to MODE 4 with RCS temperature ≤ [275°F] places the plant into a MODE where the LCO is not applicable. The 12-hour time to reach the non-applicable MODE is reasonable based on operating experience.

B.1

If the emergency power supplies to the heaters are not capable of providing [200 kW] or the pressurizer heaters are not available, restoration is required in 72 hours. The basis for 72 hours is engineering judgment that a demand caused by loss of offsite power will not occur in this period. Pressure control may be maintained during this time using normal station-powered heaters.

C.1 and C.2

If the power supplies and/or the heaters cannot be restored, power reduction to MODE 3 and then to MODE 4 places the plant in a condition where the LCO is not applicable. The time periods are reasonable based on operating experience; MODE 3 can be achieved in six hours and MODE 4 can be achieved in 12 hours from full power without challenging plant systems.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This surveillance requires pressurizer water level to be verified within the maximum limits on a periodic basis. The surveillance is performed by observing indicated level. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analysis assumptions.

SR 3.4.9.2

The surveillance is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.) The frequency of 92 days is based on industry-accepted practice.

REFERENCES

1. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

2. NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

3. Letter from T. E. Murley (USNRC) to J. K. Gasper (C-E Owners' Group) transmitting "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Group's Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," dated May 9, 1988.

SR 3.4.9.3

Th. surveillance demonstrates that the heaters can be manually transferred to and energized by emergency power supplies. The frequency of [18] months is based on a typical fuel cycle and industry accepted practice. This is consistent with similar verifications of emergency power.

16A.7.10 B 3.4.10 PRESSURIZER SAFETY VALVES

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the four spring loaded pressurizer safety valves is to provide Reactor Coolant System (RCS) overpressure protection. Operating in conjunction with the reactor protection system, four valves are used to assure that the safety limit of 2750 psia is not exceeded for analyzed transients during operation in MODES 1 and 2. Four safety valves are used for MODE 3 and portions of MODE 4. For the remainder of MODE 4 and for MODE 5, overpressure protection is provided by operating procedures and LCO 3.4.8, Low Temperature Overpressurization Protection (LTOP) System. For these conditions, ASME requirements are satisfied with one safety valve.

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The pressurizer safety valves are designed in accordance with the requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III. (Ref. 1). The required lift pressure is 2500 psia ± 1%. The safety valves discharge steam from the pressurizer to the Incontainment Refueling Water Storage Tank located in the containment.

APPLICABLE SAFETY ANALYSES

All accident analyses in CESSAR-DC which require safety valve actuation assume operation of all pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis is also based on operation of all safety valves and assumes that the valves open at the high range of the setting (2500 psia system design pressure plus 1%). These valves must accommodate pressurizer insurges which could occur during various heatup events such as rod withdrawal, ejected rod, loss of main feedwater, loss of load or main feedwater line break accident. The loss of load event with delayed reactor trip establishes the minimum safety valve capacity. The single failure of a safety valve to open is neither assumed in the accident analysis nor required to be addressed by the ASME code. Compliance with this specification is required to assure that the accident analysis and design basis calculations remain valid. The pressurizer safety valves are components that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, the pressurizer safety valves satisfy the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 2).

LCOs

The four pressurizer safety valves are set to open at the RCS design pressure (2500 psia) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure Safety Limit, to maintain accident analysis assumptions, and to comply with ASME Code requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this specification is the reactor coolant pressure boundary Safety Limit of 110% of design pressure. Inoperability of one or more valves could result in exceeding the Safety Limit were a transient to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

The note suspending LCO 3.0.4 and SR 3.0.4 permits testing and examination of the safety valves at high pressure and temperature near their normal operating range but only after the valves have had a preliminary cold setting. The cold setting gives good assurance that the valves are as close as possible to the operating setting. The note permits a pragmatic approach to ensure that the valves are OPERABLE near their design condition. Only one valve will be removed from service at a time for testing. The 72-hour exemption is based on 18-hours outage time for each of the four valves. The 18-hour period is derived from operating experience that hot testing can be performed in this time frame.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP temperature, OPERABILITY of four valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included although the listed accidents may not require all safety valves for protection. The LCO is not applicable in MODE 4 below ~~[317 °F]~~ and MODE 5 because LTOP ~~protection~~ is provided.

[259 °F] for cooldown and [290 °F] for heatup (the heatup rate is limited to 40 °F/hr or less])

Overpressure protection is not required in MODE 6 with the reactor vessel head detensioned.

BASES

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place in 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the RCS pressure boundary.

or by placing the plant in shutdown cooling with the LTOP relief valves in service in 12 hours.

B.1 ~~and~~ B.2 and B.3

If the Required Action cannot be met within the required Completion Time, the plant must be placed in a MODE in which the requirement does not apply. This is done by placing the plant in at least MODE 3 in six hours and
[259°F] in MODE 4 below [317°F] in 12 hours. The six hours allowed to reach MODE 3 is a reasonable time based on operating experience to reach MODE 3 from full power without challenging plant systems. Similarly, the 12 hours
[259°F] allowed is a reasonable time based on operating experience to reach MODE 4 without challenging plant systems. Below [317°F], overpressure protection is provided by LTOP. The change from MODES 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by four pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

Surveillance Requirements are specified in the Inservice Testing Program. Section XI of the ASME Code (Ref. 1) provides the activities and the Frequency necessary to satisfy the Surveillance Requirements. No additional requirements are specified.

REFERENCES

1. ASME Boiler & Pressure Vessel Code, Section III, "Nuclear Vessels," Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components".

2. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

16A.7.11 B 3.4.11 LOW TEMPERATURE OVERPRESSURIZATION PROTECTION (LTOP) SYSTEM

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Low Temperature Overpressurization Protection (LTOP) System

BASES

BACKGROUND

The purpose of the Low Temperature Overpressure Protection (LTOP) System LCO is to limit reactor coolant pressure at low temperatures to levels which will not compromise Reactor Coolant Pressure Boundary (RCPB) integrity (Ref. 1). The reactor vessel is the limiting component for demonstrating that protection is provided. The reactor vessel material is less tough at reduced temperatures than at normal operating temperature. As reactor vessel neutron irradiation accumulates, the vessel material becomes less resistant to stress at low temperatures (Ref. 2). Stresses are therefore maintained low and increased only as temperature increases.

Overpressure protection given by the LCO is provided ~~by ensuring that only one safety injection (SI) pump is operable, and~~ by placing the SCS relief valves in service or depressurizing the Reactor Coolant System (RCS) through an open vent. The open RCS vent or the SCS relief valves are the overpressure protection devices which provide backup to the operator in terminating increasing pressure events. ~~The approach used to protect the vessel also requires deactivating all but one SI pump because the RCS vents are not fully capable of preventing overpressurization if this system was inadvertently actuated.~~

APPLICABLE SAFETY ANALYSES

Analyses have been performed in response to NRC requests to demonstrate that the reactor vessel is adequately protected against overpressurization during shutdown. Transients potentially capable of overpressurizing the Reactor Coolant System have been identified and evaluated. Postulated transients include ~~inadvertent safety injection actuation;~~ opening safety injection tank discharge valves; energizing the pressurizer heaters; failing the makeup control valve open; temporary loss of decay heat removal; reactor coolant thermal expansion caused by reactor coolant pump (RCP) start causing heat transfer from hot steam generators; and, addition of nitrogen to the pressurizer.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LTOP system was designed to protect the RCS from overpressurization resulting from any of the following conditions:

1. The starting of an idle RCP with the secondary water temperature of the steam generator ≤ [100°F] above the RCS cold leg temperature.

2. The simultaneous starting of ~~a SI pump~~ all four SI pumps and its injection into the RCS.

During the two design bases events, no operator action is assumed to take place until ten minutes have passed.

In Reference 3, the NRC specified that the LTOP features met Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 4) and that the LCO be retained in Technical Specifications.

LCOs

The LCO requires ~~that only one SI pump be OPERABLE and~~ that the SCS Relief Valve be OPERABLE with a setpoint at the overpressure limit, with the block valve open to ensure a clear flow path, or the RCS be depressurized via an open vent.

APPLICABILITY

This LCO is applicable in MODE 4 with the temperature of any RCS cold leg < [317°F], in MODE 5, and in MODE 6 with the reactor vessel head on. The LCO is not applicable for operating conditions above the [317°F] temperature because the pressurizer safety valves are able to provide overpressure protection. With the vessel head off, there is no need for overpressure protection. The applicability is modified by a note which states that LCO 3.0.4 is not applicable. This Note is necessary to allow entry into the applicable MODE 3 without meeting the requirements of the LCO. It would not be prudent to place the plant in a condition to meet this LCO until the plant was cooled down and RCS pressure was reduced.

(continued)

BASES

ACTIONS

~~A.1~~

~~With more than one SI pump OPERABLE, actions must be initiated immediately to render all but one pump inoperable. Inadvertent operation of more than one SI pump could present a challenge to the LTOP System resulting in RCS overpressure. The completion time of "immediately" reflects the importance of preventing RCS overpressure in this plant condition.~~

~~B.1~~ A.1

With one SCS relief valve inoperable, overpressure relieving capability is reduced and restoration of the SCS relief valve in seven days is required. The other SCS Relief Valve remains OPERABLE or the RCS must be depressurized through an open vent. Either of these paths provide adequate overpressure protection. However, redundancy has been lost. The seven-day completion time in Mode 4, and 24 hour completion time in Modes 5 and 6 (per NRC GL 90-06) reflects the need to restore redundancy and also takes into consideration the other overpressure protection paths available in this condition.

~~C.1~~ B.1

If the Required Actions cannot be met within the associated completion times, the plant must be placed in a condition where an overpressure event cannot occur. This is done by depressurizing the RCS through the open Rapid Depressurization Valves. These valves provide an opening large enough to prevent pressurization of the RCS beyond LTOP limits with injection from one SI pump and one charging pump. The Completion Time of eight hours is reasonable based on the amount of time required to place the plant in this condition and the probability of an accident requiring the LTOP System during this relatively short period of time.

(continued)

ACTIONS (continued)

~~D.1~~ C.1

In the unlikely event that both SCS relief valves are inoperable and one or more rapid depressurization valves is closed, action must be taken to restore it or establish an alternate path. The completion time of "immediately" reflects the need to restore vent path capability since inadvertent or uncontrolled operation of an SI or charging pumps could cause overpressurization.

SURVEILLANCE REQUIREMENTS

~~SR 3.4.11.1~~

~~Verification that only one SI pump is OPERABLE ensures that inadvertent operation of multiple SI pumps will not create an RCS overpressure condition to challenge the LTOP System. Verification within 15 minutes prior to decreasing RCS cold leg temperatures below [317°F] ensures that the LCO requirements are satisfied prior to entering the applicable mode. Thereafter, the surveillance is required at 12-hour intervals. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analysis assumptions.~~

SR 3.4.11.~~2~~ 1

The RCS vent must be verified open for relief protection. For vent valves that are not locked open, the required Frequency is every 12 hours. For vent valves that are locked open, the required frequency is every 31 days. These frequencies have been shown by operating practices to be sufficient to regularly assess degradation and verify operation within the safety analysis assumptions.

This surveillance is modified by a Note which requires performance of this SR only when complying with Required Action C.1. This vent path is only used when in Required Action C.1, and therefore, need be performed only when in Required Action C.1.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.~~3~~ 2

Verification that the block valve is open ensures an open flow path to each required SCS Relief Valve. Surveillance is required at 12-hour intervals. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analysis assumptions.

SR 3.4.11.~~4~~ 3

Surveillance Requirement 3.4.11.4 is the performance of a SETPOINT CALIBRATION every 18 months. The SETPOINT CALIBRATION for the LTOP setpoint ensures that the SCS Relief Valves will be actuated at the appropriate RCS pressure by verifying the accuracy of the valve lift pressure. The calibration can only be performed during a shutdown. The Frequency of 18 months is based on a typical refueling cycle and industry-accepted practice.

REFERENCES

1. 10 CFR 50, Appendix G, "Fracture Toughness Requirements."

2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operation."

3. Letter from T. E. Murley (USNRC) to J. K. Gasper (C-E Owners' Group) transmitting "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Group's Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," dated May 9, 1988.

4. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

16A.7.12 B 3.4.12 RCS OPERATIONAL LEAKAGE

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 RCS Operational LEAKAGE

BASES

BACKGROUND

Limits on leakage from the Reactor Coolant Pressure Boundary (RCPB) are required to limit system operation in the presence of excessive leakage. Leakage should be limited to amounts which do not compromise safety. These leakage limits ensure appropriate action can be taken before the integrity of the RCPB is impaired.

This LCO specifies the types and amounts of leakage which are acceptable during continued plant operation. This LCO is required to limit degradation of the RCPB. The safety significance of leaks from the RCPB can vary widely depending on the source of the leak as well as the leakage rate and duration. Therefore, the detection and monitoring of reactor coolant leakage into the containment area is necessary. Separating the identified sources of leakage from unidentified sources is necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action should a leak occur that is detrimental to the safety of the public.

A limited amount of leakage is expected from auxiliary systems within the containment. If leakage occurs from these paths, it should be detected, located, and isolated from the containment atmosphere if possible, so as not to mask any potentially serious RCPB leak. This LCO protects the RCPB against continuing degradation and helps assure that serious leaks or Loss of Coolant Accidents (LOCAs) will not develop. The consequences of violating this LCO include the possibility of further degradation of the RCPB which may lead to a LOCA.

APPLICABLE SAFETY ANALYSES

Primary-Secondary LEAKAGE is a factor in the dose releases resulting from accidents or transients involving secondary steam release to the atmosphere such as a steam line break. The leak permits contamination of the secondary fluid in the steam generator. The assumption of one (1) gpm Primary-to-Secondary LEAKAGE was used as an initial condition for all non-LOCA events in Chapter 15 of Reference 1.

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APPLICABLE SAFETY ANALYSES (continued)

Primary-to-Secondary LEAKAGE is a process variable that is an initial condition of a design basis accident or transient analyses that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 2).

LCOs

a. Pressure Boundary LEAKAGE

No Pressure Boundary LEAKAGE is allowed because it would be indicative of material deterioration. Pressure Boundary LEAKAGE is defined as leakage through a non-isolable fault in an RCS component body, pipe, or vessel wall (excluding RCP shaft seals, packing, and steam generator tube leakage). Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the RCPB.

b. Unidentified LEAKAGE

One gpm of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump/holdup volume tank monitoring equipment can detect within a reasonable time period. Unidentified LEAKAGE is defined as reactor coolant leakage which is not identified. Violation of this LCO could result in continued degradation of the RCPB, if the leakage is from the pressure boundary.

c. Identified LEAKAGE

Identified LEAKAGE is defined as leakage into closed systems connected to the RCS that is captured and recovered. Up to 10 gpm of identified LEAKAGE is considered allowable because leakage is from known sources which do not interfere with detection of unidentified LEAKAGE and is well within the capability of the makeup system.

(continued)

BASES

LCOs (continued)

Identified LEAKAGE includes leakage to the containment from sources that are specifically known and located, but does not include pipe or vessel leakage or controlled RCP seal leakoff (which is a normal function and is not considered leakage). Violation of this LCO could result in continued degradation of a component or system.

d. Primary-to-Secondary LEAKAGE

Total Primary-to-Secondary LEAKAGE of one (1) gpm to all steam generators produces acceptable offsite doses in the accident analysis. Violation of this LCO could void the offsite dose calculations. Primary-to-Secondary LEAKAGE is to be included in the total allowable limit for Identified LEAKAGE.

e. Primary-to-Secondary LEAKAGE Through One Steam Generator

The 720 gallons per day (gpd) limit on one steam generator is based on allocating the total one (1) gpm allowed Primary-to-Secondary LEAKAGE equally between the two steam generators.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB leakage is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not provided because the reactor coolant pressure is far lower resulting in lower stresses and a reduced potential for leakage.

Other related LCOs include LCO 3.4.13, RCS Pressure Isolation Valve Leakage, which specifies leakage limits for certain valves that isolate the high pressure RCS from other low pressure systems and Surveillance 3.4.13.1 measures leakage through each PIV individually. Since there are two PIVs in series in each PIV line, leakage measured through one PIV may not result in any RCS LEAKAGE if the other is leak tight. If both series valves leak resulting in a loss of mass from the RCS, the loss is to be included in the

(continued)

BASES

APPLICABILITY (continued)

allowable Identified LEAKAGE. LCO 3.4.14, RCS Leakage Detection Instrumentation, specifies the requirements for the monitoring equipment used to detect leakage into the containment.

ACTIONS

A.1

With Identified LEAKAGE, Unidentified LEAKAGE, or Primary-to-Secondary LEAKAGE in excess of the LCO limits, the leakage should be reduced within four hours. This completion time allows four hours to verify leakage rates and either identify Unidentified LEAKAGE or reduce leakage to within limits, before the unit must proceed towards shutdown conditions. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any Pressure Boundary LEAKAGE exists or if Identified, Unidentified, or Primary-to-Secondary LEAKAGE cannot be reduced to within limits within four hours, the unit must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. The reactor must be placed in MODE 3 within six hours and MODE 5 within 36 hours. This action reduces the leakage and also reduces the factors which tend to degrade the pressure boundary. The completion time of six hours is reasonable based on operating experience, to reach MODE 3 from full power without challenging plant systems. Similarly, the completion time of 36 hours to reach MODE 5 is reasonable based on operating experience to reach the required MODE without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower and further deterioration is less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1

Verifying that RCS LEAKAGE is within the LCO limits assures that the integrity of the RCPB is maintained. Pressure Boundary Leakage would at first appear as Unidentified Leakage and can only be positively identified by inspection. Unidentified LEAKAGE and Identified Leakage are demonstrated to be within limits by performance of a RCS water inventory balance. Primary-to-Secondary LEAKAGE is also measured by performance of an

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

RCS water inventory balance in conjunction with effluent monitoring within the secondary Feedwater and Steam Systems. The RCS water inventory balance must be performed with the unit at steady state and near operating pressure. Therefore, when entering MODES 3 or 4, the requirements of SR 3.0.4 are not applicable for performing an RCS inventory balance.

An early warning of Pressure Boundary LEAKAGE or Unidentified LEAKAGE is provided by the automatic systems which monitor the containment atmosphere radioactivity or containment sump.

The frequency permits a reasonable interval for trending of leakage while recognizing the relative importance of early leak detection in the prevention of accidents. Steady state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful and the surveillance is not required unless steady state is established. For purposes of leakage determination by inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank level, constant makeup and letdown and reactor coolant pump seal injection and return flows. Pressure Boundary LEAKAGE would be detected more quickly by the leakage detection systems referenced in LCO 3.4.14, RCS Leakage Detection Instrumentation.

REFERENCES

1. CESSAR-DC Chapter 15, "Accident Analysis."

2. 52 FR 3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

16A.7.13 B 3.4.13 RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

This specification applies to the four series check valves (two per line) that isolate the high pressure Reactor Coolant System (RCS) from low pressure portions of the shutdown cooling (SDC) (SCS) System, outside the containment. Two valves in series are required to provide redundancy of isolation, and the concept of the LCO is to provide two barriers. A high pressure rated, motor operated gate valve is upstream of the two check valves. The selection of valves is based on information presented in Reference 1 which requires testing of two in-series check valves used for isolation of high pressure to low pressure systems when leakage of one valve could go undetected for a substantial length of time.

PIV leakage limits apply to leakage rates for individual valves. The total identified leakage rate of 10 gpm given by LCO 3.4.13 also applies, but only when a loss of RCS mass through the two series valves is indicated by an inventory balance (Ref. SR 3.4.13.1). Since there are two PIVs in series in each PIV line, leakage measured through one PIV may not result in any RCS LEAKAGE if the other is leak tight.

Although the specification provides limits in the form of allowable leakage rates, the important purpose of the specification is to prevent overpressure failure of the low pressure portions of the ~~SDC system~~ SCS caused by high RCS pressure. The leakage limits are symptoms that the boundary (check valves) between the RCS and the ~~SDC system~~ SCS is degraded or becoming degraded. Failure of the check valves could lead to overpressure of the ~~SDC~~ SCS piping or components. Failure consequences could be a Loss of Coolant Accident (LOCA) outside of containment, with the possibility of being unable to recirculate from the containment after the initial Incontainment Refueling Water Storage Tank (IRWST) injection.

The basis for this LCO is the Reactor Safety Study, WASH-1400 (Ref. 2), which identified potential intersystem LOCA as a significant contributor to plant risk. A subsequent study (Ref. 3) evaluated various PIV configurations to determine the probability of intersystem LOCA. This study determined that periodic leak testing of PIVs can reduce the probability of a LOCA.

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APPLICABLE SAFETY ANALYSES	Pressure isolation valve leakage is not considered in any design basis accident analyses. This specification provides for monitoring the condition of the reactor coolant pressure boundary to detect degradation which could lead to accidents. Therefore, Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 4) is satisfied.
LCOs	The allowable leakage for valves is based on [0.5 gpm] per inch of (nominal) valve diameter up to a maximum of five gpm. Since all valves are [10-inch] diameter nominal, the limit is [five gpm]. Violation of this LCO could result in continued degradation of a pressure isolation valve, may cause failure of the boundary between the high pressure RCS and low pressure systems, and result in loss of reactor coolant outside containment.
APPLICABILITY	In MODES 1, 2, 3, and 4, this LCO applies because the potential for PIV leakage is greatest when the RCS is pressurized. In MODES 5 and 6, leakage limits are not provided because the reactor coolant pressure is far lower resulting in a reduced potential for leakage and a lower potential for LOCA outside the containment.
ACTIONS	A.1 With leakage in excess of the allowable limits, four hours are provided to reduce leakage. The period permits operation to continue under stable conditions while leakage is assessed and corrective actions are being taken. These include actions to verify leakage and actions to reseat leaking valves. The four-hour time allows these actions to be taken and restricts the time of operation with a single isolation valve.

(continued)

BASES

ACTIONS (continued)

A.2.1 and A.2.2

If restoration is not possible, the flow path must be isolated by at least one valve (two are preferred).

Required Actions A.2.1 and A.2.2 are modified by a note to specify that the valves used for isolation must meet the same leakage requirements as the PIVs, must be in a high pressure section of piping, and must be rated for the pressure. The initial isolation (A.2.1) with one valve must be performed within four hours. This four-hour period is based on similar rationale to that of A.1. This action is inherently accomplished if only one of the two original valves is leaking because the other PIV provides the required isolation. Required Action A.2.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring the leaking PIV. The 72-hour time is based on engineering judgment and is consistent with the outage time allowed for a single division of Safety Injection.

B.1 and B.2

If leakage cannot be reduced or the system isolated, then the plant must be placed in a mode in which the requirement does not apply. This is done by placing the plant in MODE 3 within six hours and MODE 5 within 36 hours. This action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed completion times are reasonable to achieve the required MODES from full power without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Performance of leakage testing on each RCS PIV is required to verify that leakage is below the specified limits and to detect leaking valves. The leakage limit of [0.5 gpm] per inch of nominal valve diameter [5 gpm] total for each of these valves) is to be applied to each valve. Leakage assessment requires a stable plant pressure condition. Testing is performed every 18 months which is a typical refueling cycle. This Surveillance Requirement was specified by the NRC in Reference 1 and is in accordance with ASME XI (Item 3) (Ref. 5).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

In addition, testing must be performed after the valves have been opened by flow or exercised to ensure tight reseating. Testing must be performed within 24 hours after the valves have been reseated. The 24 hours is based on engineering judgment that the test is practical in this time period.

SR 3.0.4 is exempted for entry into MODES 3 and 4 to permit leak testing at high differential pressures with stable conditions not possible in the lower MODES.

REFERENCES

1. NRC Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves (all plants) dated 4/20/81. Includes Technical Evaluation Report "Primary Coolant System Pressure Isolation Valves," prepared by the Franklin Research Center.

2. USNRC, "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", Appendix V, WASH-1400 (NUREG-75/014), Oct. 1975.

3. USNRC, "The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes", NUREG-0677, May 1980.

4. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

5. ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants."

16A.7.14 B 3.4.14 RCS LEAKAGE DETECTION INSTRUMENTATION

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

NRC General Design Criteria require that means be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

A limited amount of leakage is expected from the Reactor Coolant System (RCS) and from auxiliary systems within the containment. Some leakage will occur from valve packing, pump seals, vessel/closure head seals, and safety and relief valves. If leakage occurs via these paths, it is detected, collected to the extent practical, and isolated from the containment atmosphere so as not to mask any potentially serious leak should it occur. These leakages are Identified LEAKAGE and may be piped to tanks or sumps so flow rate can be established and monitored during plant operation.

Uncollected leakage to the containment atmosphere from other sources increases the humidity of the containment. The moisture condensed from the atmosphere by air coolers together with any associated liquid leakage to the containment is Unidentified LEAKAGE and is collected in tanks or sumps and is monitored during plant operation. A small amount of Unidentified LEAKAGE may be impractical to eliminate, but it should be reduced to a small flow rate, to permit the LEAKAGE detection systems to positively and rapidly detect a small increase in flow rate. Thus a small Unidentified LEAKAGE rate that is of concern will not be masked by a larger acceptable Identified LEAKAGE rate.

Leakage detection systems should detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for a gross pressure boundary failure. Some cracks might develop and penetrate the RCPB wall, exhibit very slow growth, and afford ample time for a safe and orderly plant shutdown.

holdup volume tank (HVT) level instrument/containment sump

The leakage detection monitors used provide diversity and operate on two different principles: ~~containment sump~~ and atmospheric activity monitoring. The atmospheric activity monitoring instrumentation detects gaseous and particulate radioactivity.

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BASES

BACKGROUND (continued)

Industry practice has shown that water flow rate changes of from [0.5 to 1.0 gpm] can readily be detected in containment sumps by monitoring changes in sump water level, in flow rate, or in the operating frequency of pumps. Sumps and tanks used to collect Unidentified LEAKAGE and air cooler condensate are instrumented to alarm. This sensitivity provides an acceptable performance for detecting increases in Unidentified LEAKAGE.

INT/

Reactor coolant activity released to the containment can be detected by radiation monitoring instrumentation. Instrument sensitivities of [10^{-9} micro Ci/cc] radioactivity for air particulate monitoring and of [10^{-6} micro Ci/cc] radioactivity for gaseous monitoring are practical for these leakage detection systems.

In addition to the instrumentation cited by the LCO, other leakage detection means may be used. Humidity changes or pressure and temperature changes may provide indications of leakage.

APPLICABLE SAFETY ANALYSES

The safety significance of leaks from the RCPB can vary widely depending on the source of the leak as well as the leakage rate and duration. Therefore, the detection and monitoring of reactor coolant leakage into the containment area is necessary. Separating the identified sources of leakage from unidentified sources is necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action should a leak occur that is detrimental to the safety of the public.

RCS leakage detection instrumentation satisfies Selection Criterion 1 of the NRC Interim Policy Statement (Ref. 1). As such, these variables are retained in the RCS Leakage Detection Instrumentation LCO.

LCOs

One method of protection against RCPB leakage failure is the ability of instrumentation to rapidly detect extremely small leaks. This LCO requires that instrumentation of two diverse principles be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition when leakage indicates possible RCPB degradation.

(continued)

BASES

LCOs (continued)

The LCO is satisfied when monitors of diverse measurement means are available. Thus the HVT/ containment sump monitor in combination with either a particulate or a gaseous activity monitor provides an acceptable minimum.

APPLICABILITY

In MODES 1, 2, 3, and 4, leakage detection systems are required to be OPERABLE to support LCO 3.4.12, RCS Operational LEAKAGE. Therefore, the LCO is not applicable in MODES 5 and 6.

ACTIONS

A.1 and A.2

With the HVT/ containment sump monitor inoperable, no form of grab sample could provide the equivalent information. However, the atmospheric activity monitors provide indications of changes in leakage. Restoration is required to regain the function of the sump monitor. As an alternate to the HVT/ sump monitor and in conjunction with atmospheric monitors the periodic surveillance, SR 3.4.12.1, for RCS inventory balance is to be performed at an increased frequency of 24 hours to provide periodic information that is adequate to detect leakage. The 31-day completion time for restoration recognizes that multiple forms of leakage detection are available.

B.1.1, B.1.2 and B.2

With both types of containment atmosphere radioactivity monitoring instrumentation (gaseous and particulate activity monitor) inoperable, a water inventory balance in accordance with SR 3.4.12.1 must be performed or grab samples shall be taken and analyzed to provide periodic information. Provided the inventory balance is performed or samples are obtained and analyzed every 24 hours, the plant may continue operation for up to 31 days. The 24-hour interval provides periodic information that is adequate to detect leakage. The 31-day completion time for restoration recognizes that multiple forms of leak detection are available.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If the Required Action cannot be met within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within six hours and in MODE 5 within 36 hours. The allowed completion times are reasonable to achieve the required MODES from full power without challenging plant systems.

D.1

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Surveillance Requirement 3.4.14.1 is the performance of a CHANNEL CHECK of the containment atmosphere (gaseous and particulate) activity monitor. The CHANNEL CHECK gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analysis assumptions.

SR 3.4.14.2

Surveillance Requirement 3.4.14.2 is a performance of a CHANNEL FUNCTIONAL TEST for the containment atmosphere (gaseous and particulate) activity monitor. This test ensures that the monitor can perform its function in the desired manner. The CHANNEL FUNCTIONAL TEST verifies the alarm setpoint and relative accuracy of the instrument strings. The Frequency of 31 days is based on industry-accepted practice.

SR 3.4.14.3 ~~and~~, SR 3.4.14.4 and SR 3.4.14.5

HVT Level switch

Surveillance Requirements 3.4.14.3 and 3.4.14.4 are the performance of CHANNEL CALIBRATIONS of the containment atmosphere activity monitor and containment sump monitor every 18 months. The CHANNEL CALIBRATION verifies the accuracy of the instrument string. The calibration includes the calibration of instruments located inside containment. The frequency of 18 months is based on a typical refueling cycle and industry-accepted practice.

REFERENCES

1. 52 FR 3788, NRC Interim Policy Statement on Technical Improvements for Nuclear Power Reactors, February 6, 1987.

16A.7.15 B 3.4.15 RCS SPECIFIC ACTIVITY

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Specific Activity

BASES

BACKGROUND

The Code of Federal Regulations 10 CFR 100 (Ref. 1) specifies the maximum dose to the whole body and thyroid an individual at the site boundary can receive for two hours during an accident. The limits on specific activity ensure that the dose is held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The Design Basis events that have the greatest sensitivity to RCS specific activity are the steam line break, letdown line break, feedwater line break and steam generator tube rupture. The purpose of the Reactor Coolant System (RCS) specific activity LCO is to limit the concentration of radionuclides in the reactor coolant and the resultant offsite dose consequences of these events.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activities. The allowable levels are intended to limit the two-hour dose at the SITE BOUNDARY to be within acceptable values. The limiting values for specific activities in the LCO represent standardized limits based upon a parametric evaluation by the NRC of offsite radioactivity dose consequences for typical site locations. These evaluations showed that the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 guideline dose limits, assuming a broad range of site applicable atmospheric dispersion factors in a parametric evaluation. These standard limits on specific activity were also used in establishing standardization in shielding and unit personnel radiation protection practices.

APPLICABLE SAFETY ANALYSES

The LCO limitation on the specific activity of the reactor coolant ensures that the resulting two-hour dose at the SITE BOUNDARY will not exceed SRP dose guidelines (Ref. 1) following the accidents analyzed in Chapter 15 of CESSAR-DC. In the safety analyses, the specific activity of the reactor coolant is assumed to be at the LCO limit and an existing reactor coolant-steam generator tube leakage rate of one (1) gpm is assumed. In addition for some events a pre-existing (PIS) and event generated iodine spike is assumed.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

For accidents with a PIS, the RCS activity was assumed to be at the limit given in Action B.1 (60 μCi/gm). Operation with iodine specific activity levels greater than the LCO limit is permissible, provided that the activity levels do not exceed 60 μCi/gm and do not exist for more than 48 hours.

When specific activity exceeds the LCO limits due to iodine spiking but is limited to 60 μCi/gm, plant operation is considered acceptable based upon the low probability of an accident occurring during the established 48-hour time limit, together with the fact that iodine spiking is considered in the safety analysis.

The reactor coolant specific activity is a process variable that is an initial condition of a design basis accident that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 2).

LCOs

The specific iodine activity is limited to 1.0 microcurie per gram DOSE EQUIVALENT I-131 and the total specific activity in the reactor coolant is limited to the number of microcuries per gram equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the two-hour dose to an individual at the site boundary during design basis accidents will be limited to SRP acceptance criteria. The limit on $100/\bar{E}$ ensures the two-hour whole body dose to an individual at the site boundary during design basis accidents will also be limited to SRP acceptance criteria.

Violation of the LCO may result in reactor coolant radioactivity levels that could lead to established dose limits being violated.

(continued)

BASES

APPLICABILITY

In MODES 1, 2, and 3, operation within the LCO limits for DOSE EQUIVALENT I-131 and total specific activity are necessary to contain the potential consequences of accidents to within the acceptable SITE BOUNDARY dose values. For operation in MODES 4, 5, and 6, the probability of a steam, feedwater or letdown line break is small due to the low primary and secondary pressures and the release of radioactivity in the event of a SGTR is prevented since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves. In all applicable MODES, with the LCO limits exceeded, an isotopic analysis for iodine concentration is appropriate to monitor the activity level while actions are being taken to reduce the specific activity level.

ACTIONS

A.1 and A.2

With the gross activity in excess of the allowed limit, an analysis is to be performed within four hours to determine DOSE EQUIVALENT I-131. The Completion Time of four hours is reasonable based on the typical time to obtain, transport, and analyze a sample. If the gross activity is not reduced in this period, the plant must be placed in MODE 4. With activity levels of this magnitude present, the calculated radiological dose at the SITE BOUNDARY could exceed acceptable values. The change to MODE 4 operation lowers the saturation pressure for the reactor coolant below the setpoints of the main steam safety valves. This action prevents venting of the steam generator to the environment in the event of a SGTR. The completion time of twelve hours is reasonable based on operating experience to reach MODE 4 from full power without challenging plant systems.

ACTIONS

B.1 and B.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit of 1.0 μCi/gm, frequent samples at intervals not to exceed four hours are to be taken to demonstrate that the limit 60 μCi/gm is not exceeded. The Completion Time of four hours is reasonable based on the typical time to obtain, transport, and analyze a sample. Sampling is to continue to provide a trend. If the limit violation resulted from nominal iodine spiking, then the DOSE EQUIVALENT I-131 should be restored to nominal within 48 hours.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If the DOSE EQUIVALENT I-131 exceeds the 60 μCi/gm limit or is > 1.0 μCi/gm for a continuous time interval of 48 hours, an abnormal condition is indicated and the reactor must be placed in MODE 4 within 12 hours. The Completion Time of 12 hours is based on engineering judgment and is considered a reasonable time to get to MODE 4 from full power without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

The surveillance is performed at least once per 72 hours to monitor the gamma isotopic analysis of the reactor coolant. It basically is a quantitative measurement of radionuclides with half lives > 4 minutes, excluding radioiodines. This measurement considers the sum of the degassed gamma activity and the total of the identified gaseous activities in the sample taken. This surveillance provides an indication of any increase in the specific activity of the reactor coolant. Monitoring of the results of this surveillance allows for proper remedial actions to be taken prior to reaching the LCO limits under normal operating conditions. This surveillance is applicable in MODES 1, 2, and 3. The frequency of 72 hours has been shown to be acceptable through operating experience.

SR 3.4.15.2

This surveillance is performed to ensure iodine levels remain within limits following power changes. The 14-day Frequency is adequate to trend changes in the activity level considering that gross activity is monitored every 72 hours. The Frequency between two and six hours following a power change ≥ 15% RTP within a one-hour period is established because iodine spikes occur at this time. Samples at any other time would not provide as accurate results.

(continued)

BASES

SURVEILLANCE REQUUIREMENTS (continued)

SR 3.4.15.3

A radiochemical analysis for Ē determination is required to be performed every six months with the plant operating in MODE 1 with equilibrium conditions. These requirements for Ē determination directly relate to the LCO and are required to verify unit operation within the specified LCO limits. The radiochemical analysis for Ē is a measurement of the average energies per disintegration of isotopes with half lives > 4 minutes, excluding iodines. The Frequency of six months is based on the fact that Ē does not change rapidly during operation. This Frequency has been shown to be acceptable through operating experience.

SR 3.0.4 does not apply so that sampling can be performed in MODE 1. The sample must be taken after a minimum of two Effective Full Power Days and 20 days of power operation have lapsed since the reactor was last subcritical for ≥ 48 hours. This ensures that the activity is at equilibrium so the analysis for Ē is representative and is not skewed by a crud burst or other event.

REFERENCES

1. 10 CFR 100, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," USNRC, 1973.

2. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, USNRC, February 6, 1987.

16A.7.16 B 3.4.16 RCS LOOPS - TEST EXCEPTIONS

B 3.4 REACTOR COOLANT SYSTEMS (RCS)

B 3.4.16 RCS Loops - Test Exceptions

BASES

BACKGROUND	This special test exception permits reactor criticality at reduced RCS temperatures and under reduced flow conditions during PHYSICS TESTS while at low THERMAL POWER levels.
APPLICABLE SAFETY ANALYSES	There are no transient or accident analyses which specify the allowed boundaries of this LCO. However, operating experience has demonstrated this exception to be safe under the present applicability.
LCO	The LCO is provided to allow for the performance of PHYSICS TESTS in MODE 2. Without the LCO, plant operations would be held bound to the normal operating LCOs for reactor coolant loops and circulation (MODES 1 and 2), a minimum temperature for criticalities, and minimum pressure, temperature air flow limits. Hence, the appropriate physics tests could not be performed. In MODE 2 where the associated PHYSICS TESTS must be performed, operation is allowed under no reduced conditions provided the reactor trip setpoints of the OPERABLE power level channels are set at ≤ [5%] RTP. This ensures if some problem caused the plant to enter MODE 1 and start increasing plant power, the Reactor Protection System would automatically shut it down before power became too high, and thereby prevent violation of fuel design limits. The LCO also requires that both RCS loops and at least one RCP in each loop be in operation to provide forced coolant circulation. It also requires that the pressure/temperature relationship be maintained.
APPLICABILITY	This LCO ensures that the plant will not be operated in MODE 1 without forced circulation. It only allows testing under these conditions while in MODE 2. Therefore, no safety or fuel design limits will be violated as a result of the associated tests.

(continued)

BASES

ACTIONS

A.1

If THERMAL POWER increases to > 5% RTP, the reactor must be tripped immediately. This ensures the plant is not placed in an unanalyzed condition, and prevents exceeding the specified acceptable fuel design limits.

B.1, B.2 and B.3

If the RCS temperature and/or pressure is outside the P/T limits, the reactor must be tripped immediately. The pressure and temperature must be restored with the P/T limits. An engineering evaluation must be performed to ensure structural integrity of the RCS prior to achieving criticality.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

THERMAL POWER must be verified to be within limits once per hour. The hourly Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within the LCO limits.

SR 3.4.16.2

Within 12 hours of initiating PHYSICS TESTS, a CHANNEL FUNCTIONAL TEST must be performed on each logarithmic power level and linear power level neutron flux monitoring channel. The interval is adequate to ensure that the appropriate equipment is OPERABLE to aid in monitoring and protection of the plant during these tests.

SR 3.4.16.3

Both RCS loops and at least one RCP in each loop must be verified to be in operation. The hourly frequency has been shown by operating practice to be sufficient.

SR 3.4.16.4

The RCS temperature and pressure must be verified to be with the P/T limits. The hourly frequency has been shown by operating practice to be sufficient.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.4.16.5 The RCS temperature must be verified to be ≥ [300°F]. The hourly frequency has been shown by operating practice to be sufficient.	
REFERENCES	1.	None.

B 3.

B 3.4.17 Reactor Coolant Gas Vent System

BASES

BACKGROUND

The function of the Reactor Coolant Gas Vent System (RCGVS) is to provide a safety-grade means of venting non-condensible gases and steam from the pressurizer and the reactor vessel upper head. The RCGVS is designed to be used during all design bases events for RCS pressure control purposes when main spray and auxiliary spray systems are unavailable. The operability of at least one RCGVS path from the pressurizer, and at least one RCGVS path from the reactor vessel head to the RTD or the IWRST ensures that this functioned can be performed.

from the pressurizer and the reactor vessel

The RCGVS is a manually operated safety-grade system. It removes noncondensible gases or steam through vent lines to the RDT/IRWST. Each vent line has two pairs of parallel isolation valves which are closed during normal operation. During shutdown or transient conditions, if the operator determines that noncondensible gases have collected in the pressurizer or in the reactor vessel upper head, the operator follows the operating procedures to vent the gases by manually opening the RCGVS valves from the main control room. The RCGVS will have the capability to be manually actuated, monitored, and controlled from the control room, as required by GDC 19.

The two valves in each parallel path are powered by an emergency (battery backed) ac power source, and a normal (deisel backed) ac power source. A FMEA (Table 6.7-3) demonstrates that the RCGVS will maintain a vent path after a single failure of any single valve or its power source. This demonstration satisfies the requirements of GDC 17 and GDC 34.

APPLICABLE
SAFETY ANALYSIS

The RCGVS provides a safety grade method of RCS depressurization that is credited during natural circulation and during steam generator tube rupture events. The operator uses the SI system, the pressurizer backup heaters, and the RCGVS to control RCS inventory and subcooling.

The pressurizer vent line is 1½ inch nominal diameter to meet the EPRI URD requirement to vent one-half the RCS volume in one hour. The reactor vessel vent line is a three-quarter inch line which expands to one inch through the valving. This provides adequate venting to remove steam and non-condensible gases from the reactor vessel head.

LCO

The LCO requires the RCGVS to be OPERABLE for all design basis events. The RCGVS is OPERABLE when a vent path can be established from the pressurizer and from the reactor vessel to the RDT or IRWST.

APPLICABILITY

In MODES 1, 2, 3, and 4, the two vent paths are required to be OPERABLE. The RCGVS is primarily used for natural circulation and for tube rupture events, however, it must be available for all design basis events.

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

With inoperable components, such that at least one valve train in the vent path from the reactor vessel upper head to the RTD or the IRWST is not OPERABLE, at least one of the two valve trains must be returned to OPERABLE status within 72 hours. If at least one RCGVS valve train in the vent path from the reactor vessel to the RDT/IRWST cannot be made OPERABLE within 72 hours, then the plant must be in MODE 3 within an additional 6 hours, and then in MODE 5 within an additional 36 hours.

B.1, B.2, and B.3

With inoperable components, such that at least one valve train in the vent path from the pressurizer to the RDT/IRWST is not operable, at least one of the two valve trains must be returned to OPERABLE status within 72 hours. If at least one RCGVS valve train in the vent path from the pressurizer to the RDT cannot be made OPERABLE within 72 hours, then the plant must be in MODE 3 within an additional 6 hours, and then in MODE 5 within an additional 36 hours.

~~function. If at least one RCGVS vent path from the pressurizer to the RDT cannot be made OPERABLE within 72 hours, then the plant must be in MODE 3 within an additional 6 hours, and then to MODE 5 within an additional 36 hours.~~

<u>C1, C2, and C3</u>

With components inoperable, such that none of the RCGVS vent paths are OPERABLE, at least one of the RCGVS vent paths must be returned to OPERABLE status within 6 hours. ~~In this condition, the RDT will be capable of performing the depressurization function.~~ If at least one RCGVS vent path cannot be made OPERABLE within 6 hours, then the plant must be in MODE 3 within an additional 6 hours, and then in MODE 5 within an additional 36 hours.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

There is one manual valve in the RCGVS; it is in the vent path from the reactor vessel upper head. It is necessary to verify that this valve is locked open to ensure that a vent path can be established from the reactor vessel upper head to the RDT. The 18 month frequency is based on accessibility during the refueling cycle.

SR 3.4.17.2

Cycling each vent valve through at least one complete cycle verifies the RCGVS valves will function when necessary. The frequency of 18 months is based on a typical refueling cycle, and is an industry accepted practice. ~~for valves in the primary pressure boundary.~~

SR 3.4.17.3

Verifying that the pressure instrument root valves are open ensures that line pressure between valves can be monitored. The 18 month frequency is based on accessibility during the refueling cycle.

SR 3.4.17.4

Verifying flow through the vent paths when cycling the valves (SR 3.4.17.2) ensures the RCGVS vent paths are OPERABLE. The frequency of 18 months is based on the frequency of valve cycling tests, and the refueling cycle frequency.

SR 3.4.17.5

Verification of correct breaker alignment and valve position indications ensures that valves can be operated when required,

and valve position can be monitored. The frequency of seven days is accepted industry practice and has been shown to be acceptable by operating experience.

<INSERT B>

B. One valve in a vent path found open	B.1 Restore valve to closed position OR B.2 Be in MODE 3 OR B.3 Be in MODE 5	12 hours 6 hours 36 hours

B3

B3. Rapid Depressurization Function

BASES

BACKGROUND

The Rapid Depressurization Function (RDF) of the Safety Depressurization System (SDS) is designed as a manually operated safety-grade system that removes steam or water from the pressurizer through two isolation valves in each of two parallel depressurization lines to the Incontainment Refueling Water Storage Tank. The RDF is designed to mitigate the consequences of a beyond-design-basis event such as a total loss of normal and emergency feedwater (TLOFW).

The RDF valves are closed during normal operation. These valves are motor operated and fail in the "as is" position. The emergency power dc busses supply electrical power to the motor operators. Table 6.7-3 (FMEA) demonstrates that an RDF bleed path can be established in the event of a single failure of the valves or the battery banks with total station blackout. This design feature satisfies GDC 17 and GDC 34. The two globe valves in the RDF have their power removed during normal operation to prevent inadvertantly opening a vent path. The RDF will have the capability to be manually actuated, monitored, and controlled from the control room, as required by GDC 19.

The RDF also performs an important function in mitigating a severe accident. During a core melt, the system would allow the RCS to be depressurized and reduce the possibility of a challenge to the containment, such as from direct containment heating.

APPLICABLE
SAFETY ANALYSIS

The design-basis event for determining the size of the RDF bleed valves is a TLOFW event. The analysis was performed using a realistic version of the CEFLASH-4AS code with assumed best estimate decay heat values. Use of the realistic version of the CEFLASH-4AS code is acceptable because the RDF is designed to mitigate accidents beyond the design basis. Letdown, charging, and pressurizer spray were not credited. In the accident scenario, the applicant assumes the initial RCS power and secondary steam are generated at the rated output. The primary and secondary valves open at lift pressures of 2500 psia and 1200 psia, respectively, and the RCS pumps trip 10 minutes after the event is initiated.

Two cases were analyzed: (1) a TLOFW event with one RDF bleed path open, two SI pumps operable, and immediate operator action to open the RDF bleed path after the primary safety valves (PSVs) open, and (2) a TLOFW event with both RDF bleed paths operable, four SI pumps operable, and an operator action delay of 30 minutes to open the RDF paths after the PSVs open. The analysis shows that case 2 is the worst case, which requires larger RDF bleed valves, each sized to meet the acceptance criteria.

LCO

The LCO requires the RDF to be OPERABLE. Both vent paths shall be closed for all design basis events. The RDF is OPERABLE when a vent path can be established from the pressurizer to the IRWST.

APPLICABILITY

In MODES 1, 2, 3, and 4, at least one vent path is required to be OPERABLE, and both vent paths closed. The RDF is for use in beyond-design-bases events such as a TLOFW, and for mitigating severe accidents such as a core melt.

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

With inoperable components, such that at least one vent path is not OPERABLE, at least one vent path must be returned to OPERABLE status within 72 hours. If at least one RDF vent path cannot be made OPERABLE within 72 hours, then the plant must be in MODE 3 within an additional 6 hours, and then in MODE 5 within an additional 36 hours. The 72 hour completion time is based on the extremely low probability of the beyond-design-basis event (TLOFW) that the RDF is designed for.

ACTIONS B.1, B.2, and B.3

ANSI 51.1 requires two closed valves to maintain the pressure boundary. If one valve is found open, at any time except when the RDF is being used for its design functions, the valve must be closed within 12 hours. Otherwise, the plant must be MODE 3 within an additional 6 hours, and be in MODE 5 within an additional 36 hours.

SURVEILLANCE
REQUIREMENTS

SR 3.4.18.1

Verifying that the pressure instrument root valves are open ensures that line pressure between the globe and gate valves can be monitored. The 18 month frequency is based on accessibility during the refueling cycle.

SR 3.4.18.2

Cycling each vent valve through at least one complete cycle verifies the RDF valves will function when necessary. The frequency of 18 months is based on a typical refueling cycle, and is an industry accepted practice.

SR 3.4.18.3

Periodic verification of the correct valve position indication in the control room for all RDF valves ensures that the valves are properly aligned, and that the position indicators are functioning properly. A frequency of seven days is accepted by industry practice, and has been shown to be acceptable by operating experience.

SR 3.4.18.4

Verification of correct breaker alignment and power availability to the valve indicators ensures that valves can be operated when required, that vent paths will not be inadvertently opened, and valve position can be monitored. Breakers are normally ON for the gate valves and OFF for the globe valves. The frequency of seven days is accepted industry practice, and has been shown to be acceptable by operating experience.

16A.8 B 3.5 SAFETY INJECTION SYSTEM (SIS)

16A.8.1 B 3.5.1 SAFETY INJECTION TANKS

B 3.5 SAFETY INJECTION SYSTEM (SIS)

B 3.5.1 Safety Injection Tanks

BASES

BACKGROUND

The functions of the four Safety Injection Tanks (SITs) are to supply water to remove heat from the core during the blowdown phase of a Loss of Coolant Accident (LOCA), and to provide a water inventory to aid in refilling the core and reactor vessel downcomer during the recovery phase that follows. The SITs are passive components since no operator or control actions are required for them to perform their function. Internal tank pressure and gravity are sufficient to discharge the tank contents to the Reactor Coolant System (RCS) if RCS pressure decreases below SIT pressure.

Each SIT discharges its water volume directly to the reactor vessel downcomer via a direct vessel injection nozzle, also utilized by the Safety Injection System. Each SIT is isolated from the RCS by a motor operated isolation valve and two check valves in series. The motor operated isolation valves are normally open with power removed from the valve motor to prevent inadvertent closure prior to, or during an accident. Additionally, they are interlocked with the pressurizer pressure measurement channels to ensure the valves will automatically open as RCS pressure is increased above SIT pressure, and to prevent inadvertent closure prior to an accident. The valves also receive a Safety Injection Actuation Signal (SIAS) to open. These features ensure the valves meet the requirements of IEEE Std 279-1971 (Ref. 1) for "operating bypasses" and that the SITs will be available for injection without reliance on operator action.

The SIT gas/water volumes, gas pressure, and outlet pipe size are selected to allow three of the four SITs to partially recover the core before significant clad melting or zirconium-water reaction can occur following a LOCA. The need to ensure that three SITs are adequate for this function is consistent with LOCA analysis assumption that the entire contents of one SIT will be lost via the break during the blowdown phase of a LOCA.

(continued)

BASES

BACKGROUND (continued)

This LCO helps to ensure that the following acceptance criteria established by 10 CFR 50.46 (Ref. 2) for Emergency Core Cooling Systems (ECCS) willbe met following a LOCA:

- Maximum fuel element cladding temperature of ≤ 2200°F.
- Maximum cladding oxidation of ≤ 0.17 times the total cladding thickness before oxidation.
- Maximum hydrogen generation from a zirconium-water reaction of ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
- The core is maintained in a coolable geometry.

Since the SITS discharge during the blowdown phase of a LOCA, they do not contribute to the long term requirements of 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The SITs are credited for in both the large and small break LOCA analysis at full power, (Ref. 3). In performing the LOCA calculations, conservative assumptions are made concerning the availability of safety injection flow. In the early stages of a LOCA with a loss of off-site power, the SITs provide the sole source of makeup water to the RCS. This is because the safety injection pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg breaks, the entire contents of one SIT is assumed to be lost through the break during blowdown, even through the SITs discharge their contents directly to vessel downcomer via the direct vessel injection nozzle.

The limiting large break LOCA is a Double Ended Guillotine Cold Leg break at the discharge of the reactor coolant pump. During this event the SITs discharge to the RCS as soon as RCS pressure decreases below SIT pressure. As a conservative estimate in the calculation of the reflood portion of the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

accident, no credit is taken for safety injection pump flow until the SITs empty. This results in a minimum effective delay of over [60] seconds during which the SITs must provide the core cooling function. The actual delay time does not exceed 40 seconds. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA assumes a time delay of 140 seconds before pumped flow reaches the core. The SITs and an SI pump both play a part in terminating the rise in clad temperature. As break size decreases the role of the SITs decreases until they are not required and the SI pumps become solely responsible for terminating the temperature increase.

Since the SITs are passive components, active single failures are not applicable to their operation. The SIT isolation valves, however, are not single failure proof; therefore, whenever the valves are open, power shall be removed from their operators. These precautions ensure that the SITs are available during an accident (Refs. 4, 5). With power supplied to the valves a single active failure could result in a valve closure, which would render one SIT unavailable for injection. If a second SIT is lost through a DVI break only two SITs would reach the core. Since the only active failure which could affect the SITs would be the closure of a motor-operated outlet valve, the requirement to remove power from these eliminates this failure mode.

The minimum volume requirement for the SITs ensures that three SITs can provide adequate inventory to reflood the core and downcomer following a LOCA. The downcomer then remains flooded until the Safety Injection Pumps start to deliver flow.

The maximum volume limit is based upon maintaining an adequate gas volume to ensure proper injection and the ability of the SITs to fully discharge, and to limit the maximum amount of boron inventory in the SITs. A minimum of [ ] narrow range level corresponding to [1600] cubic feet and a maximum of [ ] narrow range level corresponding to [1927] cubic feet of borated water are used in the safety analysis as the volume in the SITs. To allow for instrument accuracy, [ ] narrow range corresponding to [1625] cubic feet and [ ] narrow range corresponding to [1902] cubic feet, are specified. The analysis is based upon the cubic feet requirements, the % figures are provided for operator use since the level indication provided in the control room is in %, not cubic feet.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The minimum pressure requirement ensures that the contained gas volume will generate discharge flow rates during injection which are consistent with those assumed in the safety analysis.

The maximum pressure limit ensures excessive amounts of gas will not be injected into the RCS after the SITs have emptied.

A minimum pressure of [570] psig and a maximum pressure of [632] psig are used in the analysis. To allow for instrument accuracy, [575] psig minimum and [627] psig maximum are specified.

The maximum allowable boron concentration in the SITs is based upon boron precipation limits in the core following a LOCA. Establishing a maximum limit for boron is necessary since the time at which boron precipation would occur in the core following a LOCA is a function of break location, break size, the amount of boron injected into the core and the point of ECCS injection. Post-LOCA emergency procedures directing the operator to establish simultaneous hot/cold leg injection are based upon the worst case minimum boron precipation time. Maintaining the maximum SIT boron concentration within the upper limit ensures the SITs do not invalidate this calculation. An excessive boron concentration in any of the borated water sources used for injection during a LOCA could result in boron precipation earlier than predicted.

The minimum boron requirements are based on Beginning Of Life reactivity values and are selected to ensure the reactor will remain subcritical during the reflood stage of a large break LOCA. During a large break LOCA all Control Element Assemblies (CEAs) are assumed not to insert into the core and the initial reactor shutdown is accomplished by void formation during blowdown. Sufficient boron concentration must be maintained in the SITs to prevent a return to criticality during reflood. Although this requirement is similar to the basis for the minimum boron concentration of the In-Containment Refueling Water Storage Tank (IRWST) the minimum SIT concentration is lower than the IRWST since the SITs need not account for dilution by the RCS.

The SITs satisfy the requirements of Criterion 3 of the Interim Policy Statement as described in Reference 6.

(continued)

BASES

LCO

The LCO establishes the minimum conditions required to ensure the SITs are available to accomplish their core cooling safety function following a LOCA. Four SITs are required OPERABLE to ensure 100% of the contents of three of the SITs will reach the core during a LOCA. This is consistent with the assumption that the contents of one tank spill through the break for a DVI line break. If less than three tanks are injected during the blowdown phase of a LOCA the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) may not be satisfied.For a SIT to be considered OPERABLE, the isolation valve must be fully open with power removed and the limits established for contained volume, boron concentration and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and ~~MODE 3~~ MODES 3 and 4 with RCS pressure ≥ [900] psia the SIT OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases the SITs are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures ≥ [900] psia. Below [900] psia, the rate of RCS blowdown is such that the SI Pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.

In MODE 3 ≤ [900] psia and in MODES 4, 5 and 6, the SIT motor-operated isolation valves are closed to isolate the SITs from the RCS. This allows RCS cooldown and depressurization without discharging the SITs into the RCS or requiring depressurization of the SITs.

ACTIONS

A.1

With the boron concentration of one SIT not within limits, the boron concentration must be returned to within the limits within 72 hours. In this condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. However, since the volume of the SIT is still available for injection and the boron requirements are based on the average boron concentration of the total volume of three SITs, the consequences are less severe than those with a SIT not available for injection. The 72-hour

(continued)

BASES

ACTIONS

A.1 (continued)

Completion Time to return the boron concentration to within limits is consistent with the requirements imposed on other Engineered Safety Features systems in similar MODES.

B.1

With one SIT inoperable, for a reason other than boron concentration, the SIT must be returned to OPERABLE status within one hour. In this condition the required contents of three SITs cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the one hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover-pressure ensures prompt action is taken to return the inoperable SIT to OPERABLE status. The Completion Time minimizes the time exposure of the plant to a LOCA in these conditions.

C.1

The plant must be placed in a MODE in which the LCO does not apply if the SIT cannot be returned to OPERABLE status within the associated Completion Time. This is done by placing the plant in at least MODE 3 in six hours and by reducing pressurizer pressure to < [900] psia within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power without challenging plant systems.

D.1

With more than one SIT inoperable, the plant is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification of proper valve position, as indicated in the Control Room, ensures the SITs are available for injection and ensures timely discovery if

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1 (continued)

a valve should be less than fully open. If an isolation valve is not fully open the rate of injection to the RCS can be reduced. Although a motor-operated valve position should not change with power removed, a closed valve could result in not meeting accident analysis assumptions. A 12-hour frequency ensures a mispositioned isolation valve will be quickly identified while limiting the time the plant would be operated in a degraded condition.

SR 3.5.1.2 and 3.5.1.3

This surveillance ensures that the nitrogen cover-pressure and the borated water volume contained in the SITs are sufficient to ensure adequate injection during a LOCA. Due to the static design of the SITs, a 12 hour frequency allows the operator to identify changes before the limits are reached and has been shown to be acceptable through operating experience.

SR 3.5.1.4

This surveillance ensures SIT boron concentration is within the required limits. Due to the static design of the SITs, a 31-day frequency is adequate to identify changes which could occur from mechanisms such as stratification or in-leakage. Sampling within six hours after a 1% volume increase will identify if in-leakage from the RCS has caused a reduction in boron concentration below the required limit.

SR 3.5.1.5

This surveillance ensures that an active failure could not result in the closure of a SIT motor-operated isolation valve coincident with a LOCA. If this were to occur only two SITs would be available for injection assuming one SIT contents are lost. Installation and removal of the breakers is conducted under administrative control. Since this surveillance is a verification that the breaker is removed and is relatively easy, the 31-day frequency was chosen to provide additional assurances that the breakers are removed.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.5 (continued)

This SR is modified by a Note which allows power to be supplied to the motor-operated isolation valves when RCS pressure is < [900] psia, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves. Were closure to occur, in spite of the interlock, the SIAS signal provided to the valve would open a closed valve should a LOCA occur.

REFERENCES

1. IEEE Std. 279-1971, Criteria For Protection Systems for Nuclear Power Generating Stations.
2. 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants.
3. CESSAR-DC Section 6.3, "Safety Injection System."
4. Branch Technical Position BTP ICSB - 4, attached to SRP Requirements on Motor-Operated Valves in the ECCS Accumulator Lines.
5. Branch Technical Position BTP ICSB - 18, attached to SRP Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves.
6. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-335, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application)."
7. Draft NUREG-1366, "Improvements to Technical Specification Requirements."

(continued)

BASES

REFERENCES (continued)

Additional References

8. 10 CFR 50, Appendix A, GDC 35 - Emergency Core Cooling System.

9. 10 CFR 50, Appendix A, GDC 36 - Inspection of Emergency Core Cooling System.

10. 10 CFR 50, Appendix A, GDC 37 - Testing of Emergency Core Cooling System.

11. 10 CFR 50, Appendix K - ECCS Evaluation Models.

12. GL 85-16, High Boron Concentrations, August 23, 1985.

13. RG 1.79, Preoperational Testing of Emergency Cooling Systems for Pressurized Water Reactors, Rev. 01, September, 1975.

14. SRP 6.3, Emergency Core Cooling Systems, April, 1984.

15. NRC memorandum R. L. Bayer to V. Stello, Jr., recommended Interim Revisions to LCOs for ECCS Components, December 1, 1975.

16A.8.2 B 3.5.2 SIS - OPERATING

B 3.5 SAFETY INJECTION SYSTEM (SIS)

B 3.5.2 SIS - Operating

BASES

BACKGROUND

The function of the SIS is to supply water to remove decay heat from the core, both short and long term, in the event that normal cooling systems (steam generators or shutdown cooling) cannot provide this function and to supply borated water to the reactor core following increased heat removal events, such as large steam line breaks, and to provide inventory for a S/G tube rupture event.

Four redundant 100% capacity safety injection (SI) pumps are provided. In MODES 1, 2 and 3, all divisions are required. This ensures that 100% of the core cooling requirements can be provided even in the event of a single DVI line break with a failure of DG to start.

An independent suction header supplies water from the Incontainment Refueling Water Storage Tank to each of the safety injection pumps. Each SI pump discharges directly to the reactor vessel downcomer via the Direct Vessel Injection nozzle. The SI pump flow directs sufficient flow to the core to meet the analysis assumptions following a loss of coolant accident (LOCA) in one of the RCS cold legs.

During low temperature conditions in the RCS, limitations are placed on the maximum number of SI Pumps which may be OPERABLE. Refer to the Bases for LCO 3.4.11, Low Temperature Overpressure Protection (LTOP) System, for the basis of these requirements.

LCO 3.5.2 helps to ensure that the following acceptance criteria established by 10 CFR 50.46 (Ref. 1) for SIS will be met following a LOCA:

- o Maximum Fuel element cladding temperature of $\leq$ 2200°F
- o Maximum cladding oxidation of $\leq$ 0.17 times the total cladding thickness before oxidation.

(continued)

BASES

BACKGROUND (continued)

- o Maximum hydrogen generation from a zirconium-water reaction of ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

- o The core is maintained in a coolable geometry.

- o Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post-trip return to power following a steamline break event and a CEA ejection accident.

During a large break LOCA RCS pressure will decrease to less than 200 psia in less than 20 seconds. The safety injection systems are actuated upon receipt of a SIAS. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available the safeguard loads start immediately in the programmed sequence. If offsite power is not available the Engineered Safety Features buses shed normally operating loads and are connected to the emergency diesel generators. Safeguards loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading and pump starting determine the time required before pumped flow is available to the core following a LOCA.

APPLICABLE SAFETY ANALYSES

The SI system is assumed to be OPERABLE in the large break LOCA analysis at full power, CESSAR-DC 6.3 (Ref. 2). This analysis establishes a minimum required runout flow for the SI pumps. The SI Pumps are credited in the small break LOCA analysis. This analysis establishes the flow and discharge head requirements at the design point for the SI Pump. The main steam line break event also establishes the flow-head requirement and in addition establishes the minimum required response time for actuation of the pumps. The Steam Generator Tube Rupture (STGR) CEA ejection and inadvertent opening of an atmospheric dump valve analyses also credit the SI Pumps, but do not limit the design.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The large break LOCA event with a loss of offsite power and a single failure (disabling two SIS divisions) establishes the OPERABILITY requirements for the SIS. During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or CEA insertion for small breaks. Following depressurization, emergency cooling water is injected into the direct vessel injection nozzles, flows down the downcomer, fills the lower plenum, and refloods the core.

On smaller breaks, RCS pressure will stabilize at a value dependent upon break size, heat load and injection flow. The smaller the break, the higher this equilibrium pressure. In all LOCA analyses injection flow is not credited until RCS pressure drops below the shutoff head of the SI Pumps.

The LCO ensures an SIS division will deliver sufficient water to match decay heat boil off rates soon enough to minimize core uncovery for a large LOCA. It also ensures that the SI Pump will deliver sufficient water during a small LOCA, and provide sufficient boron to maintain the core subcritical following a SLB.

The SIS divisions satisfy the requirements of Criterion 3 of the Interim Policy Statement as described in Reference 3.

LCO

In MODES 1, 2 and 3, four independent (and redundant) SIS division are required to ensure sufficient SIS flow is available to mitigate the consequences of a LOCA assuming a single failure coincident with a LOOP. Additionally, the SIS divisions may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2 and 3, an SIS division consists of a SI pump, the piping, instruments and controls to ensure an OPERABLE flow path capable of taking suction from the IRWST on a SIAS.

(continued)

BASES

LCO (continued)

During an event requiring SIS actuation, a flowpath is provided to ensure an abundant supply of water from the IRWST to the RCS via the SI pumps and their respective supply lines to each of the four direct vessel injection nozzles. In the long term, flowpaths may be switched to supply part of its flow to the RCS hot legs via the hot leg injection nozzles on two of the divisions.

The flowpath for each division must maintain its designed independence to ensure that no single failure can disable more than two SIS divisions.

APPLICABILITY

In MODES 1, 2, and 3 the SIS OPERABILITY requirements for the limiting DBA, large LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The SI Pump performance is based on the small break LOCA which establishes the pump performance curve and has less dependence on power. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis. The steam line break event must also be considered in developing the pump performance curve.

MODE 4 SIS functional requirements are described in LCO 3.5.3.

In MODES 5 and 6, plant conditions are such that an event requiring SIS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, Reactor Coolant Loops and Circulation Loops Filled, and LCO 3.4.8, Loops Not Filled. MODE 6 core cooling requirements are addressed by LCO 3.9.4, Shutdown Cooling and Coolant Circulation, High Water Level, and LCO 3.9.5, Shutdown Cooling and Coolant Circulation, Low Water Level.

(continued)

BASES

ACTIONS

A.1

With one or more components inoperable such that 100% of the equivalent flow of an SIS division is not fully OPERABLE, the inoperable components must be returned to OPERABLE status within 72 hours. In this condition, the remaining OPERABLE SIS components are adequate to perform the SIS function. However, the overall reliability is reduced because a single failure in the remaining OPERABLE components could result in reduced SIS capability. The 72-hour Completion Time is based on NRC recommendations (Ref. 4) based on a risk evaluation, and is a reasonable time for many repairs.

An SIS flowpath is inoperable if it is not capable of delivering the design flow to the RCS. The individual components are inoperable if they not capable of performing their design function, or supporting systems are not available (except as allowed by their respective LCOs).

The LCO requires the OPERABILITY of a number of independent subsystems. An event accompanied by a loss of offsite power and the failure of an emergency diesel generator can disable two SIS divisions until power is restored. It is assumed that flow from the third SI pump is discharged through the break. Analysis has shown that flow from one SI pump is sufficient to keep the core covered for a DVI line break which is the limiting SBLOCA. Hence, continued operation for 72 hours is justified. In Condition B, the plant is assured of having more equipment available than in Condition A, and the same Completion Time for these two Conditions is justified.

B.1 and B.2

The plant must be placed in a MODE in which the LCO does not apply if the inoperable components cannot be returned to OPERABLE status within the associated Completion Time. This is done by placing the plant in at least MODE 3 in six hours ~~and by reducing pressurizer pressure to < [1700 psia]~~ followed by placing the plant in Mode 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power without challenging plant systems.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures the flowpath from the SIS pumps to the RCS is maintained. Misalignment of these valves could render the associated SIS division inoperable. Securing these valves in position by removal of power ensures they cannot change position as the result of an active failure or be inadvertently misaligned. These valves are of the type described in Reference 5 that can disable the function of the associated SIS divisions, invalidating the accident analysis. A 12-hour frequency ensures a mispositioned valve will be quickly identified while limiting the time the plant would be operated in a degraded condition.

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the SIS flowpaths provides assurance that the proper flowpaths will exist for SIS pump. This SR does not apply to valves which are locked, sealed or otherwise secured in position since these were verified to be in the correct position prior to locking, sealing or securing. A valve which receives an actuation signal is allowed to be in a non-accident position provided the valve will automatically reposition within the proper stroke time. This Required Action 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. The 31-day frequency is appropriate because the valves are operated under procedural control, an improper valve position would only affect a single train, and the probability of an event requiring SIS actuation during this time period is low. This frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

With the exception of systems in operation, the SIS pumps are normally in a standby, non-operating mode. As such, flowpath piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the SIS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of non-condensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.3

following an SIAS or during shutdown cooling. The 31-day frequency is based on the low probability of an event requiring SIS actuation during this time, the gradual nature of gas accumulation in the SIS piping, and the procedural controls governing system operation.

SR 3.5.2.4

Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. A quarterly frequency for such tests is a Code requirement.

SR 3.5.2.5 and SR 3.5.2.6

These SRs demonstrate each automatic SIS valve actuates to its required position on an actual or simulated Safety Injection Actuation Signal (SIAS) and that each SIS Pump starts on receipt of an actual or simulated SIAS. The 18-month frequency was developed considering it is prudent that these surveillances only be performed during a plant outage. This is due to the plant conditions needed to perform the SRs and the potential for unplanned plant transients if the SRs are performed with the reactor at power. Although the actuation logic is tested as part of the ESFAS functional test every 92 days, and equipment performance is monitored as part of the Inservice Testing Program. Operating experience has shown that these components virtually always pass the SR when performed on the 18-month frequency which is consistent with the refueling cycle.

SR 3.5.2.7

Periodic inspections of the SIS suction inlet and the IRWST Holdup Volume Tank ensures that it is unrestricted and it stays in proper operating condition. A frequency of 18 months is sufficient to detect abnormal degradation and is consistent with refueling cycle.

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BASES

REFERENCES

1. 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants.

2. CESSAR-DC Section 6.3, "Safety Injection System."

3. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-335, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application)."

4. NRC Memorandum R. L. Bayer to V. Stello, Jr., Recommended Interim Revisions to LCOs for ECCS Components, December 1, 1975.

5. IE Information Notice No. 87-01, RHR Valve Misalignment Causes Degradation of ECCS in PWRs, January 6, 1987.

Additional References

6. 10 CFR 50, Appendix A, GDC 35 - Emergency Core Cooling System.

7. 10 CFR 50, Appendix A, GDC 36 - Inspection of Emergency Core Cooling System.

8. 10 CFR 50, Appendix A, GDC 37 - Testing of Emergency Core Cooling System.

9. 10 CFR 50, Appendix K - ECCS Evaluation Models.

10. GL 85-16, High Boron Concentrations, August 23, 1985.

11. GL 85-22, Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage.

12. RG 1.1, Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps, Rev. 0, November, 1980.

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BASES

REFERENCES (continued)

13. BTP MTEB 6-1, pH for Emergency Coolant Water for PWRs.

14. NUREG-0869, Containment Emergency Sump Performance, October 1985.

15. RG 1.79, Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors, Rev. 01, September 1980.

16. RG 1.82, Sumps for Emergency Core Cooling and Containment Spray Systems, Rev. 01, November 1985.

17. SRP 6.3, Emergency Core Cooling Systems, April 1984.

16A.8.3 B 3.5.3 SIS - SHUTDOWN

B 3.5 SAFETY INJECTION SYSTEM (SIS)

B 3.5.3 SIS - Shutdown

BASES

BACKGROUND	The Background section for Bases B 3.5.2 is applicable to this Bases with the following modifications: In MODE 4 an SIS division is defined as one SI subsystem. In this MODE the decay heat generation and RCS blowdown rates are such that a single SI pump is capable of providing the core cooling function in the event of a Loss of Coolant Accident (LOCA). Also in MODE 4, a zero power steam line break will have negligible consequences with respect to a reactivity transient.
APPLICABLE SAFETY ANALYSES	The Applicable Safety Analysis section of Bases 3.5.2 is applicable to this Bases.
LCO	In Mode 4 with RCS cold leg temperature >317°F two operable SI divisions ensures at least one pump is capable of adequate flow to the CORE in the event of a LOCA at a DVI line. With RCS cold leg temperature ≤ [317°F], a maximum of one SI pump is allowed to be OPERABLE in accordance with LCO 3.4.11, Low Temperature Overpressure Protection System.
APPLICABILITY	In Mode 4, with RCS temperature >317°F a loss of coolant resulting from a DVI line break requires two separate SI divisions be operable to ensure that if a LOCA disables one division an alternate SIS division is available. The requirement of having two OPERABLE SI division are acceptable without single failure consideration on the basis of the stable reactivity condition and the limited core cooling requirements. In MODES 5 and 6, plant conditions are such that an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.~~6~~7, RCS Loops -MODE 5, Loops Filled and LCO 3.4.~~7~~8, RCS MODE 5, Loops Not Filled. MODE 6 core cooling

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BASES

APPLICABILITY (continued)

requirements are addressed by LCO 3.9.4, Shutdown Cooling and Coolant Circulation, High Water Level, and LCO 3.9.5, Shutdown Cooling and Coolant Circulation, Low Water Level.

ACTIONS

A.1

With only one SI Pump OPERABLE, the unit is not prepared to respond to a LOCA. The one hour Completion Time to restore at least two SIS division to OPERABLE status ensures prompt action is taken to provide the required cooling capacity or to initiate actions to place the unit in MODE 5 where two SIS divisions are not required.

B.1

The plant must be placed in a MODE in which the LCO does not apply if two SIS Pumps cannot be returned to OPERABLE status within the associated Completion Time. This is done by placing the plant in MODE 5 in 24 hours. The 24-hour Completion Time limits the time the plant is subject to this event and is a reasonable time, based on the stable reactivity conditions of the reactor and the limited core cooling requirements, to reach conditions where this LCO is no longer applicable.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable surveillance descriptions from Bases LCO 3.5.2 apply.

REFERENCES

1. 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants.

2. CESSAR-DC Section 6.3, "Safety Injection System."

3. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-335, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application)."

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BASES

REFERENCES (continued)

4. NRC Memorandum R. L. Bayer to V. Stello, Jr., Recommended Interim Revisions to LCOs for ECCS Components, December 1, 1975.

5. IE Information Notice No. 87-01, RHR Valve Misalignment Causes Degradation of ECCS in PWRs, January 6, 1987.

Additional References

6. 10 CFR 50, Appendix A, GDC 35 - Emergency Core Cooling System.

7. 10 CFR 50, Appendix A, GDC 36 - Inspection of Emergency Core Cooling System.

8. 10 CFR 50, Appendix A, GDC 37 - Testing of Emergency Core Cooling System.

9. 10 CFR 50, Appendix K - ECCS Evaluation Models.

10. GL 85-16, High Boron Concentrations, August 23, 1985.

11. GL 85-22, Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage.

12. RG 1.1, Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps, Rev. 0, November, 1980.

13. BTP MTEB 6-1, pH for Emergency Coolant Water for PWRs.

14. NUREG-0869, Containment Emergency Sump Performance, October 1985.

15. RG 1.79, Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors, Rev. 01, September 1980.

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BASES

REFERENCES (continued)

16. RG 1.82, Sumps for Emergency Core Cooling and Containment Spray Systems, Rev. 01, November 1985.

17. SRP 6.3, Emergency Core Cooling Systems, April 1984.

16A.8.4 B 3.5.4 IN-CONTAINMENT REFUELING STORAGE WATER TANK (IRWST)

B 3.5 SAFETY INJECTION SYSTEM (SIS)

B 3.5.4 In-Containment Refueling Storage Water Tank (IRWST)

BASES

BACKGROUND

The In-containment Refueling Water Storage Tank (IRWST) supports the SIS by providing a source of borated water for Engineering Safety Feature (ESF) pump operation.

The IRWST supplies four divisions of SIS. Each SIS division is supplied by a separate, suction line. The IRWST also supplies 2 divisions of containment spray and 2 shutdown cooling pumps. Use of a single IRWST to supply four divisions of SIS and two divisions of containment spray is acceptable since the IRWST is a passive component and passive failures are not assumed in the injection phase of an accident.

The Safety Injection (SI), Shutdown Cooling (SC) Pumps and Containment Spray (CS) pumps are provided with recirculation lines which ensure each pump can maintain minimum flow requirements when operating at shutoff head conditions. These lines discharge back to the IRWST.

This LCO ensures that the IRWST contains sufficient borated water to support the SIS during the injection phase, ensures the reactor remains subcritical following a LOCA, and ensuress that the assumptions used in the safety analysis for containment net free volume are maintained. Insufficient water inventory in the IRWST could result in insufficient cooling capacity of the SIS. Improper boron concentrations could result in loss of SHUTDOWN MARGIN or excessive boric acid precipitation in the core following a LOCA.

Storage capacity of the IRWST is based on operational and safety guards requirements. The location of the SIS suction piping in IRWST will result in some portion of the stored volume being unavailable for injection. The minimum LCO volume is based upon SIS. The maximum LCO volume is based on containment free volume requirements. The IRWST temperature requirements are based on an inadvertent containment spray actuation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (CONTINUED)

The IRWST supplies the containment spray system covered in LCO 3.~~5~~6.6 and the SIS divisions, covered in LCOs 3.5.~~1~~2 and 3.5.~~2~~3, with the abundant supply of cooling water to meet GDC 35, Ref. 3.

During accident conditions the IRWST provides a source of borated water to the SI and CS pumps. As such, it provides containment cooling and depressurization, core cooling and replacement inventory, RCS depressurization using feed and bleed methods, and is a source of negative reactivity for reactor shutdown. The design basis transients and applicable safety analysis concerning each of these systems are discussed in the Applicable Safety Analysis section of Specifications 3.5.2 SIS Divisions - Operating, 3.5.3 SIS Divisions - Shutdown, and 3.6.6 Containment Spray System.

This LCO establishes the minimum requirements for contained volume, boron concentration and temperature of the IRWST inventory. This ensures an adequate supply of cool, borated water is available to: cool and depressurize the containment in the event of a LOCA or steam/feed line break, cool and cover the core in the event of a LOCA, ensure the reactor remains subcritical following a LOCA or steam line break, and depressurize the RCS using feed and bleed methods.

The Safety Analyses assumes a minimum volume in the IRWST of [495,000 gallons]. The Safety Analysis also assumes a maximum volume of [545,800] gallons to allow for adequate containment free volume. In addition the IRWST Holdup Volume Tank is assumed dry. To allow for instrument accuracy a minimum volume of [505,000 gallons] and a maximum volume of [535,000 gallons] is specified.

The limit established for minimum boron concentration is based upon ensuring that with a minimum IRWST level following a LOCA, the reactor will remain subcritical in the cold condition following mixing of the IRWST and RCS water volumes. Small break/LOCAs assumes all control rods inserted except the most reactive assembly which is withdrawn from the core. Large break LOCAs assume all CEAs remain withdrawn from the core. The most limiting case occurs at beginning of life.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The maximum boron limit in the IRWST is based on boron precipitation in the core following a LOCA. With the reactor vessel at saturated conditions, the core dissipates heat by pool nucleate boiling. Because of this boiling phenomenon in the core, the boric acid concentration will increase in this region. If allowed to proceed in this manner a point will be reached where boron precipitation will occur in the core. Post-LOCA emergency procedures direct the operator to establish simultaneous hot/cold leg injection to prevent this condition by establishing a forced flow path through the core regardless of break location. These procedures are based upon the minimum time at which precipitation could occur assuming maximum boron concentrations exist in the borated water sources used for injection following a LOCA. Boron concentrations in the IRWST in excess of the limit could result in precipitation earlier than assumed in the analysis.

The limits on IRWST temperature are determined by an inadvertent containment spray actuation. Relatively cold containment spray water will reduce the containment atmosphere temperature and thus the air and steam partial pressures. Since the air pressure in the shield building isn't immediately affected, this would put a negative pressure across the containment vessel. The design pressure is -2.0 psig.

The final containment pressure after an inadvertent containment spray actuation is sensitive to the initial containment atmosphere temperature and the IRWST water temperature. Figure 3.5.4-1 shows the minimum allowed IRWST water temperature for a given containment atmosphere temperature. For example, if the containment atmosphere temperature is 90°F, the minimum allowed IRWST water temperature is 47°F. The maximum temperature of the IRWST is 110°F.

The IRWST meets Criterion 3 for inclusion as a technical specification as it is on the primary success path for core heat removal and containment depressurization/cooling as described in the above referenced specifications.

LCO

The IRWST ensures an adequate supply of borated water is available to: cool and depressurize the containment in the event of a DBA, cool and cover the core in the event of a LOCA, ensure the reactor remains subcritical following a DBA.

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LCO (continued)	To be considered OPERABLE the limits established for water volume, boron concentration and temperature must be met.
APPLICABILITY	In MODES 1, 2, 3 and 4 the IRWST OPERABILITY requirements are dictated by the SIS and Containment Spray OPERABILITY requirements. Since both the ECCS and Containment Spray Systems must be OPERABLE in MODES 1, 2, 3 and 4, the IRWST must be OPERABLE to support their operation. In MODES 5 and 6, plant conditions are such that an event requiring SIS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCOs 3.4.7, Reactor Coolant Loops and Circulation - Loops Filled; and 3.4.8, Reactor Coolant Loops and Circulation Loops Not Filled. MODE 6 core cooling requirements are addressed by LCOs 3.9.4, Shutdown Cooling and Coolant Circulation - High Water Level; and 3.9.5. - Shutdown Cooling and Coolant Circulation - Low Water Level.
ACTIONS	A.1 With IRWST borated water volume not within limits, volume must be returned to within limits within one hour. In this condition neither the SIS or Containment Spray Systems can perform their design functions. Under these conditions prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which these systems are not required. The one-hour time limit to restore the IRWST to OPERABLE is based on this condition simultaneously affecting multiple trains. B.1 With IRWST boron concentration or borated water temperature not within limits, temperature must be returned to within limits within eight hours. In this condition neither the SIS or CS systems can perform their design functions. Under these conditions prompt action must be taken to restore the tank to OPERABLE or to place the plant in a MODE in which these systems are not required. The eight-hour limit to restore the IRWST temperature to

(continued)

BASES

ACTIONS (continued)

B.1

within limits was developed considering the time required to change boron concentration or temperature and that the contents of the tank are still available for injection.

C.1 and C.2

The plant must be placed in a MODE in which the LCO does not apply if the IRWST cannot be returned to OPERABLE status within the associated Completion Time. This is done by placing the plant in at least MODE 3 in six hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable, based on operating experience to reach the required MODES from full power operation without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The IRWST borated water temperature must be maintained within the limits assumed in the accident analysis. The 24 hour frequency is short enough to identify a temperature change that would approach either temperature limit and has been shown to be acceptable through operating experience.

SR 3.5.4.2

The IRWST water volume must be maintained above the required minimum level. If the level is too low it would not provide a sufficient initial supply for injection or enough to support continued ESF pump operation on recirculation. Since the IRWST volume is normally stable and provided with a low level alarm a seven-day frequency is appropriate and has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.3

The IRWST water volume and the IRWST Holdup Volume Tank must be at or below the stated limits to ensure that the containment free volume assumed in the safety analysis exists. Since neither tank is used in any operational mode and both are provided with alarms, a seven day frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.4

The boron concentration of the IRWST must be maintained within the required band. Maintaining the concentration within this band ensures the reactor remains subcritical following a LOCA and that the resulting sump pH is maintained in a acceptable range such that boron precipitation in the core will not occur earlier than predicted and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the IRWST volume is normally stable a seven-day sampling frequency is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-335, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application)."

2. 10 CFR 50, Appendix A, GDC 35 - Emergency Core Cooling System.

3. 10 CFR 50, Appendix A, GDC 36 - Inspection of Emergency Core Cooling System.

4. 10 CFR 50, Appendix A, GDC 37 - Testing of Emergency Core Cooling System.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Trisodium Phosphate (TSP)

BASES

BACKGROUND

Trisodium phosphate (TSP) dodecahydrate is placed ~~on the floor or in the sump~~ in the Holdup Volume Tank of the containment building to assure that iodine, which may be dissolved in the recirculated reactor cooling water following a loss of coolant accident (LOCA), remains in solution. TSP also helps inhibit stress corrosion cracking (SCC) of austenitic stainless steel components in containment during the recirculation phase following an accident.

Fuel that is damaged during a LOCA will release iodine in several chemical forms to the reactor coolant and to the containment atmosphere. A portion of the iodine in the containment atmosphere is washed to the sump by containment sprays. The emergency core cooling water is borated for reactivity control This borated water causes the sump solution to be acidic. In a low pH (acidic) solution, dissolved iodine will be converted to a volatile form. The volatile iodine will evolve out of solution into the containment atmosphere, significantly increasing the levels of airborne iodine. The increased levels of airborne iodine in containment contribute to the radiological releases and increase the consequences from the accident due to containment atmosphere leakage.

After a LOCA, the components of the core cooling and containment spray systems will be exposed to high temperature borated water. Prolonged exposure to the core cooling water combined with stresses imposed on the components can cause SCC. The SCC is a function of stress, oxygen and chloride concentrations, pH, temperature, and alloy composition of the components. High temperatures and low pH, which would be present after a LOCA, tend to promote SCC. This can lead to the failure of necessary safety systems or components.

Adjusting the pH of the recirculation solution to levels above 7.0 prevents a significant fraction of the dissolved iodine from converting to a volatile form. The higher pH thus decreases the level of airborne iodine in containment and reduces the radiological consequences from containment atmosphere leakage following a LOCA. Maintaining the

(continued)

BASES

BACKGROUND (continued)

solution pH above 7.0 also reduces occurrence SCC of austenitic stainless steel components in containment. Reducing SCC reduces the probability of failure of components.

Granular TSP dodecahydrate is employed as a passive form of pH control for post LOCA containment spray and core cooling water. Baskets of TSP are ~~placed on the floor or in the sump of~~ attached to the Primary Shield Wall of the Holdup Vol. Tank in the containment building to dissolve ~~from~~ with released reactor coolant water and containment sprays after a LOCA. Recirculation of the water for core cooling and containment sprays then provides mixing to achieve a uniform solution pH. The dodecahydrate form of TSP is used because of the high humidity in the containment building during normal operation. Since the TSP is hydrated, it is less likely to absorb large amounts of water from the humid atmosphere and will undergo less physical and chemical change than the anhydrous form of TSP.

APPLICABLE SAFETY ANALYSIS

The LOCA radiological consequences analysis takes credit for iodine retention in the sump solution based on the recirculation water pH being ≥ 7.0. The radionuclide releases from the containment atmosphere and the consequences of a LOCA would be increased if the pH of the recirculation water is not adjusted to 7.0 or above.

LCO

The TSP is required to adjust the pH of the recirculation water > 7.0 after a LOCA. A pH > 7.0 is necessary to prevent significant amounts of iodine released from fuel failures and dissolved in the recirculation water from converting to a volatile form and evolving into the containment atmosphere. Higher levels of airborne iodine in containment may increase the release of radionuclides and the consequences of the accident. A pH > 7.0 is also necessary to prevent SCC of austenitic stainless steel components in containment. SCC increases the probability of failure of components.

The required amount of TSP is based upon the extreme cases of water volume and pH possible in the containment sump after a large break LOCA. The minimum required volume is the volume of TSP that will achieve a sump solution pH of

(continued)

BASES

LCO (continued)

≥ 7.0 when taking into consideration the maximum possible sump water volume and the minimum possible pH. The amount of TSP needed in the containment building is based on the mass of TSP required to achieve the desired pH. However, a required volume is specified, rather than mass, since it is not feasible to weigh the entire amount of TSP in containment. The minimum required volume is based on the manufactured density of TSP dodecahydrate. Since TSP can have a tendency to agglomerate from high humidity in the containment building, the density may increase and the volume decrease during normal plant operation. Due to possible agglomeration and increase in density, estimating the minimum volume of TSP in containment is conservative with respect to achieving a minimum required pH.

APPLICABILITY

In MODES 1, 2, and 3, the RCS is at elevated temperature and pressure, providing an energy potential for a LOCA. The potential for a LOCA results in a need for the ability to control the pH of the recirculated coolant.

In MODES 4, 5, and 6, the potential for a LOCA is reduced or nonexistent, and TSP is not required.

ACTIONS

A.1

If it is discovered that the TSP in the containment building sump is not within limits, Action must be taken to restore the TSP to within limits. During plant operation the containment sump is not accessible and corrections may not be possible.

The Completion Time of 72 hours is allowed for restoring the TSP within limits, where possible, because 72 hours is the same time allowed for restoration of other ECCS components.

B.1 and B.2

If the TSP cannot be restored within limits within the Completion Time of Action A.1, the plant must be brought to a MODE in which the LCO does not apply. The specified Completion Times for reaching MODES 3 and 4 are those used

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

throughout the Technical Specifications; they were chosen to allow reaching the specified conditions from full power in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

ADD NEW INSERT "A" (see next page)

SR 3.5.5.1

Periodic determination of the volume of TSP in containment must be performed due to the possibility of leaking valves and components in the containment building that could cause dissolution of the TSP during normal operation. A Frequency of 18 months is required to determine visually that a minimum of [ ] cubic feet is contained in the TSP baskets. This requirement ensures that there is an adequate volume of TSP to adjust the pH of the post LOCA sump solution to a value ≥ 7.0.

The periodic verification is required every 18 months, since access to the TSP baskets is only feasible during outages, and normal fuel cycles are scheduled for 18 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of TSP placed in the containment building.

SR 3.5.5.2

Testing must be performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. A representative sample of [ ] grams of TSP from one of the baskets in containment is submerged in 1.0 ± 0.05 gallons of water at a boron concentration of [ ] ppm and at the standard temperature of 25 ± 5°C. Without agitation, the solution pH should be raised to ≥ 7 within 4 hours. The representative sample weight is based on the minimum required TSP weight of [ ] kilograms, which at manufactured density corresponds to the minimum volume of [ ] cubic ft, and maximum possible post LOCA sump volume of [ ] gallons, normalized to buffer a 1.0 gallon sample. The boron concentration of the test water is representative of the maximum possible boron concentration corresponding to the maximum possible post LOCA sump volume. Agitation of the test solution is

(continued)

6.5.2.6.3 Plant Shutdown (Startup)

The Shutdown Cooling System (SCS) is used in conjunction with the Main Steam and Emergency Feedwater Systems to reduce the Reactor Coolant System temperature in post shutdown periods from normal operating temperature to refueling temperature. In the event that one or both of the shutdown cooling pumps is unable to perform its function and the CS pumps are not required to be aligned for containment spray operation, one or both of the containment spray pumps may be aligned for shutdown cooling. This is accomplished by repositioning valves SI-104, SI-105, SI-110, SI-111, SI-430, SI-431, SI-687, and SI-695; see Figures 6.3.2-1A and 6.3.2-1B. E I

6.5.3 DESIGN EVALUATION

The CSS uses a nozzle that provides a drop size distribution which has been established by testing and found suitable for the fission product removal function. The CSS provides a nozzle pressure differential of 40 psid which fixes the drop size distribution. The mean drop size produced at this differential pressure is 530 microns. E

The CSS is designed to provide coverage for 90% of the containment net free volume. The remaining 10% of the containment net free volume is assumed to be unsprayed. The transfer rate from the unsprayed region to the sprayed region is two volumes of unsprayed region per hour.

The borated containment spray solution contains no additive for pH control during the initial stage of a LOCA. The effectiveness of the CSS in removing elemental iodine from the containment atmosphere during a LOCA is discussed in Section 15.6.5, and the calculated spray removal constant for iodine is given in Table 15A-9 of Appendix 15A.

For post-accident iodine control and to minimize corrosion of the stainless steel in the containment, the pH of the water in the IRWST and thus of the recirculated containment spray solution, is maintained at a minimum of 7.0. Disodium phosphate stored in baskets in the IRWST holdup volume becomes immersed in water during a LOCA and the resulting solution overflows into the IRWST. The stainless steel baskets, which are attached to the primary shield wall of the holdup volume, have a solid top and bottom with mesh sides to permit submergence of the disodium phosphate. The elevation of the baskets is above the normal operating water level in the holdup volume and below the IRWST spillway. Access is provided to the baskets for inspection and sampling. I

ADD TO SR 3.5.5.1 1ST Para

INSERT "A"

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.5.2 (continued)

prohibited, since an adequate standard for the agitation intensity cannot be specified. The test time of 4 hours is necessary to allow time for the dissolved TSP to naturally diffuse through the sample solution. In the post LOCA containment sump, rapid mixing would occur, significantly decreasing the actual amount of time before the required pH is achieved. This would ensure compliance with the Standard Review Plan requirement of a pH $\geq$ 7.0 by the onset of recirculation after a LOCA.

REFERENCES

1. Standard Review Plan Section 6.1.1.
2. Standard Review Plan Section 6.5.2.

16A.9 B 3.6 CONTAINMENT SYSTEMS

16A.9.1 B 3.6.1 CONTAINMENT

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment vessel, including all its penetrations, is a low leakage steel shell which is designed to withstand the postulated Loss of Coolant Accident (LOCA) or a Main Steam Line Break (MSLB) and while limiting the postulated release of radioactive material to within the requirements of 10 CFR 100 (Ref. 1). Additionally, the containment and shield building provide shielding from the fission products which may be present in the containment atmosphere following accident conditions.

The containment vessel is a 200-ft. diameter spherical steel shell with a wall thickness of approximately one and three-quarter inches. This containment shell is supported by, but not anchored to, a spherical depression in an intermediate floor of the shield building. The shield building is a reinforced concrete cylindrical building with a hemi-spherical dome which totally encloses the containment.

The internal structure is a group of reinforced concrete structures that enclose the reactor vessel and primary system. The internal structure provides biological shielding for the containment interior. The internal structure concrete base rests inside the lower portion of the containment vessel sphere.

The primary shield wall encloses the reactor vessel and provides protection for the vessel from internal missiles. The primary shield wall provides biological shielding and is designed to withstand the temperatures and pressures following LOCA. In addition, the primary shield wall provides structural support for the reactor vessel. The primary shield wall is a minimum of six feet thick.

The secondary shield wall (crane wall) provides supports for the polar crane and protects the steel containment vessel from internal missiles. In addition to providing biological shielding for the coolant loop and equipment, the crane wall also provides structural support for pipe supports/restraints and

(continued)

BASES

BACKGROUND (continued)

platforms at various levels. The crane wall is a right cylinder with an inside diameter of 130 feet and a height of 118 feet from its base. The crane wall is a minimum of four feet thick.

Containment piping penetration assemblies provide for the passage of process, service, sampling and instrumentation pipe lines into the containment vessel while maintaining containment integrity. The shield building provides biological shielding and controlled release of the annulus atmosphere under accident conditions, and environmental missile protection for the containment vessel and Nuclear Steam Supply System.

APPLICABLE SAFETY ANALYSES

The containment OPERABLE LCO was derived from the requirements related to the control of offsite radiation doses resulting from major accidents. This LCO is intended to ensure that offsite dose limits are not exceeded, by verifying that the actual containment leak rate does not exceed the value assumed in the plant safety analysis.

The Design Basis Accidents (DBAs) which result in a release of radioactive material within containment are a Loss Of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), a Main Feedwater Line Break (MFLB), and a Control Element Assembly (CEA) ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that the containment and containment shield building are OPERABLE at event initiation such that the majority of the release of fission products to the environment is controlled by the rate of containment leakage. In addition, for the above accident it is assumed that the containment low purge is operating.

The containment has been limited to an allowable leakage rate of ~~[0.34]~~ 0.5 percent of the containment volume per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J (Ref. 4) as La: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (Pa) following a DBA. The calculated maximum peak containment pressure [48.3] psig was obtained from a [0%] power MSLB DBA. The containment internal design pressure is ~~[49.0]~~ 53 psig. The allowable leakage rate represented by La forms the basis for the acceptance criteria imposed on all containment leak rate testing. Satisfactory leak test results are a requirement for the establishment of containment OPERABILITY.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The acceptance criteria applied to accidental releases of radioactive material to the environment are given in terms of total radiation dose received by a member of the general public who remains at the exclusion area boundary for two hours following onset of the postulated fission product release. The limits established in Reference 1 are a whole body dose of 25 Rem or a 300 Rem dose to the thyroid from iodine exposure, or both.

Containment OPERABLE satisfies the requirements of Selection Criteria 3 of the NRC Interim Policy Statement as documented in Reference 5.

LCO

The containment OPERABLE LCO requires the existence of a leak tight containment structure.

The provisions of this LCO are implemented by assuring:

1. Containment leakage rate are within limits.

2. Structural integrity of the containment is maintained.

The measures implemented to meet the above requirements provide assurance that the containment will perform its designed safety function to mitigate the consequences of accidents which could result in offsite exposures comparable to the Reference 2 limits.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced because of the Reactor Coolant System (RCS) pressure and temperature limitations of these MODES. In MODE 6, fuel handling evolutions are conducted. The requirements for containment during MODE 6 refueling operations are addressed in LCO 3.9.3, containment Penetrations.

(continued)

BASES

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within one hour. The one hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods where containment is inoperable is minimal.

B.1 and B.2

The plant must be placed in a MODE in which the LCO does not apply if containment cannot be restored to OPERABLE status in the associated Completion Time. This is done by placing the plant in at least MODE 3 in six hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining containment OPERABLE requires compliance with the visual examinations and leak test requirements of 10 CFR 50, Appendix J. This SR reflects the leak rate testing requirements with regard to overall containment leakage (Type A leak tests), and equipment hatch leakage (Type B leak tests) and Containment Isolation Valve, except [24-inch] purge valve (Type C leak tests). These periodic testing requirements verify that the containment leak rate does not exceed the leak rate assumed in the accident analysis. Personnel lock door seal leakage testing is addressed in LCO 3.6.2. The surveillance frequency is required by Appendix J, as such, SR 3.0.2 (which allows surveillance frequency extensions) does not apply.

REFERENCES

1. 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. System 80+ CESSAR-DC, Chapter 15, Accident Analysis.

(continued)

BASES

REFERENCES (continued)

3. System 80+ CESSAR-DC, Section 6.2, Containment Systems.

4. 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

5. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application)."

Additional References

6. 10 CFR 50, Appendix K, "ECCS Evaluation Models."

7. 10 CFR 50, Appendix A, GDC 50, "Containment Design Basis."

8. 10 CFR 50, Appendix A, GDC 52, "Capability for Containment Leakage Rate Testing."

9. 10 CFR 50, Appendix A, GDC 53, "Provisions for Containment Inspection and Testing."

10. BTP CSB 6-3 "Determination of Bypass Leakage Paths in Dual Containment."

16A.9.2 B 3.6.2 CONTAINMENT PERSONNEL LOCKS

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Personnel Locks

BASES

BACKGROUND

Two Containment Personnel Locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each personnel lock is nominally a right circular cylinder [ten] feet in diameter with a door at both ends. The doors are interlocked to prevent simultaneous opening. During periods when containment operability is not required, the door interlock mechanism may be disabled, allowing both doors to remain open for extended periods when frequent containment entry is necessary. Each personnel lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a DBA in containment. As such, closure of a single door assures the containment is OPERABLE. Each of the doors contains double gasket seals and local leakage rate testing capability to provide pressure integrity. To effect a leak tight seal, the personnel lock design uses pressure seated doors (e.g., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel lock is provided with limit switches on both doors that provide control room indication of door position. Additionally, control room indication is provided to alert the operator whenever ~~an~~ personnel lock door interlock mechanism is defeated.

The Containment Personnel Locks form part of the containment pressure boundary. As such, air lock integrity and air-tightness is essential to limit offsite doses from a Design Basis Accident (DBA). Not maintaining personnel lock integrity or air-tightness may result in offsite doses in excess of those described in the plant accident analysis.

(continued)

Amendment I
December 21, 1990

BASES

APPLICABLE SAFETY ANALYSES

The Containment Personnel Lock LCO is derived from the requirements related to the control of off-site radiation doses from major accidents by verifying that the actual containment leak rate does not exceed the value assumed in the accident analysis.

The DBAs which result in a release of radioactive material within containment are a LOCA, a Main Steam Line Break (MSLB), a Main Feed Line Break (MFLB), and a CEA ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE at event initiation, such that release of fission products to the environment is controlled by the rate of containment leakage. In addition, for the above accidents it is assumed that the containment low purge is operating. The containment has been limited to an allowable leakage rate of ~~[0.34]~~ percent 0.5
of containment volume per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J as La: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (Pa) following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the surveillance requirements associated with the personnel lock.

The acceptance criteria applied to DBA releases of radioactive material to the environment are given in terms of total radiation dose received by a member of the general public who remains at the exclusion area boundary for two hours following onset of the postulated fission product release. The limit established in Reference 1 are a whole body dose of 25 Rem or a 300 Rem dose to the thyroid from iodine exposure, or both.

Application of single failure criteria to the personnel locks is not required because the personnel locks fulfill their design safety function in a passive manner and are not subject to active failures. Therefore, closure of a single door in each personnel lock is sufficient to ensure OPERABILITY following postulated events. Nevertheless, both doors are kept closed when the personnel lock is not being used for normal entry and exit from containment.

The Containment Personnel Locks satisfy the requirements of Selection Criteria 3 of the NRC Interim Policy Statement as documented in Reference 4.

(continued)

BASES

LCO

Each Containment Personnel Lock forms part of the containment pressure boundary. As part of containment, the personnel lock safety function is related to control of offsite radiation exposures resulting from a DBA. Thus, each personnel lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each personnel lock is required to be OPERABLE. For a personnel lock to be considered OPERABLE, the interlock mechanism must be OPERABLE, the personnel lock must be in compliance with the Type B personnel lock leakage test and both doors must be OPERABLE. The interlock allows only one personnel lock door of a personnel lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. ~~The closure of a single door in an air lock will maintain containment integrity, since e~~Each door is designed to withstand the peak containment pressure calculated to occur following a DBA.

This LCO assures that the Containment Personnel Locks will perform their design~~ed~~ safety function to mitigate the consequences of accidents which could result in offsite exposures comparable to the Reference 2 limits.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the Containment Personnel Locks are not required in MODE 5 to prevent leakage of radioactive material from containment. In MODE 6, fuel handling evolutions are conducted. The requirements for the Containment Air Locks during MODE 6 refueling operations are addressed in LCO 3.9.3, Containment Penetrations. In MODES 5 & 6 with reduced RCS inventory conditions, the requirements of the Containment Air Locks are addressed in LCO 3.10.5

ACTIONS

A.1, A.2.1, A.2.2.1, and A.2.2.2

With one personnel lock door inoperable, or with one personnel lock door and interlock mechanism inoperable in one or more Containment Personnel Locks, the OPERABLE door in each affected Containment Personnel Lock must be closed and maintained closed. This assures a leak tight containment barrier is maintained by the use of an OPERABLE personnel lock door. This

(continued)

BASES

ACTIONS (continued)

A.1, A.2.1, A.2.2.1, and A.2.2.2

action must be completed within one hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1 which requires containment to be restored to OPERABLE status in one hour.

In addition, the inoperable door or inoperable interlock mechanism in each affected personnel lock must be restored to OPERABLE status or the affected personnel lock penetration must be isolated by the use of an OPERABLE personnel lock door. One of these two Required Actions must be completed within the 24 hour Completion Time. The associated Completion Time is considered reasonable for restoring the personnel lock door to OPERABLE status considering the OPERABLE door of the affected personnel lock is being maintained closed and the time required to restore the personnel lock door to OPERABLE status.

Required Action A.2.2.2 verifies that a personnel lock with an inoperable door or an inoperable door and interlock mechanism has been isolated by use of a locked and closed OPERABLE personnel lock door. This ensures that an acceptable containment leakage boundary is maintained. The periodic interval of 31 days is based on engineering judgment considering the size of the personnel lock penetration and the low probability of a locked personnel lock door being mispositioned.

The Required Actions for Condition A are modified by a note which allows entry and exit to perform repairs on the affected personnel lock component. If ALARA conditions permit, entry and exit should be via an OPERABLE Personnel lock.

B.1, B.2 and B.3

With a personnel lock door interlock mechanism inoperable in one or more personnel locks, the Required Actions and associated Completion Times consistent with Condition A are applicable.

The Required Actions for Condition B are modified by a note which allows entry and exit through the personnel lock when a dedicated individual is stationed at the personnel lock to ensure only one door is opened at any time.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

With one or more personnel lock(s) inoperable for reasons other than described in Conditions A or B, one door in the Containment Personnel Lock must be verified to be closed. This action must be completed within the one hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, Containment, which requires that containment be restored to OPERABLE status in one hour.

Additionally, the affected personnel lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered the time required for restoring an inoperable personnel lock to OPERABLE status and the relative importance of maintaining containment.

The Required Actions for Condition C are modified by a note which requires the containment to be declared inoperable should both doors in a personnel lock fail the personnel lock door seal leak test, SR 3.6.2.1. While containment integrity may be maintained in this condition (i.e., overall containment leakage rates within limits), the time required to evaluate this is considered too long to allow continued operation. Therefore, containment is declared inoperable in accordance with LCO 3.6.1.

The Required Actions for Condition C are further modified by a note which allows entry and exit to perform repairs on the affected personnel lock component. If ALARA conditions permit, entry and exit should be via an OPERABLE personnel lock.

D.1 and D.2

The plant must be placed in a MODE in which the LCO does not apply if the inoperable Containment Personnel Lock cannot be restored within the associated Completion Time. This is done by placing the plant in at least MODE 3 in six hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining Containment Personnel Locks OPERABLE requires compliance with the leak rate test requirements of 10 CFR 50, Appendix J as modified by approved exemptions. This SR reflects the leak rate testing requirements with regard to personnel lock leakage (Type B leak tests). The periodic testing requirements verify that the personnel lock leakage contribution does not result in an overall containment leak rate in excess of leak rate assumed in the safety analysis. The surveillance frequency is required by Appendix J as modified by approved exemptions. Thus, SR 3.0.2 (which allows surveillance frequency extensions) does not apply.

This SR is modified by a note to indicate an inoperable personnel lock door does not invalidate the previous successful performance of an overall personnel lock leakage test. This is considered reasonable since either personnel lock door is capable of providing a fission product barrier in the event of a DBA.

SR 3.6.2.2

The personnel lock door interlock is designed to prevent simultaneous opening of both doors in a single personnel lock. Since both the inner and outer doors of an personnel lock are designed to withstand the maximum expected post-accident containment pressure [48.3 psig], closure of either door will maintain full containment integrity. Thus, the door interlock feature ensures that containment integrity is maintained while the personnel 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 only challenged when containment is entered, this test is performed prior to entering containment but is not required more frequently than 184 days.

REFERENCES

1. 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance."

2. System 80+ CESSAR-DC, Chapter 15, Accident Analysis.

(continued)

BASES

REFERENCES (continued)

3. System 80+ CESSAR-DC, Section 6.2, Containment Systems.

4. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director, NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application)."

Additional References

5. 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

6. 10 CFR 50, Appendix K, "ECCS Evaluation Models."

7. 10 CFR 50, Appendix A, GDC 50, "Containment Design Basis."

8. 10 CFR 50, Appendix A, GDC 52, "Capability for Containment Leakage Rate Testing."

9. 10 CFR 50, Appendix A, GDC 53, "Provisions for Containment Inspection and Testing."

10. BTP CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment."

16A.9.3 B 3.6.3 CONTAINMENT ISOLATION VALVES

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment structure serves to contain radioactive material which may be released from the reactor core following a Design Basis Accident (DBA), such that offsite radiation exposures are maintained within the requirements of 10 CFR 100 (Ref. 1). In order to minimize containment leakage (and as a result, offsite radiation exposure) fluid penetrations not serving accident consequence limiting systems are provided with two isolation barriers which are closed on a Containment Isolation Actuation Signal (CIAS). Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation (and possible loss of containment integrity) or leakage that exceeds limits assumed in the accident analysis. One of these barriers may be a closed system inside containment (in accordance with the requirements of 10 CFR 50, Appendix A, Criterion 57). These barriers (typically Containment Isolation Valves) make up the containment isolation system.

Automatic Containment isolation occurs upon receipt of a high containment pressure or Safety Injection Actuation signal (SIAS). The ~~containment isolation signal~~ CIAS closes automatic Containment Isolation Valves in fluid penetrations not required for operation of engineered safeguards systems in order to prevent leakage of radioactive material. Other penetrations are isolated by the use of valves in the closed position or blind flanges. As a result, the Containment Isolation Valves (and blind flanges) help ensure that the containment atmosphere will be isolated in the event of a release of radioactive material to containment atmosphere from the RCS following a DBA. OPERABILITY of the Containment Isolation Valves (and blind flanges) ensures containment integrity is maintained during accident conditions.

~~The containment Purge Ventilation System is part of the Containment Cooling and Ventilation System. The purge system was designed for intermittent operation, providing a means of removing airborne radioactivity caused by minor reactor coolant system leakage prior to personnel entry into containment. The Containment Ventilation Purge System consists of two~~

(continued)

BASES

BACKGROUND (continued)

subsystems: a high purge and exhaust system and a low purge and exhaust system. The high purge system contains a [30,000 cfm] supply fan and a [30,000 cfm] exhaust fan while the low purge system contains a [1250 cfm] supply fan and a [1250 cfm] exhaust fan. The high and low purge supply and exhaust lines are each supplied with inside and outside Containment isolation valves.

The high purge system is designed to purge the containment atmosphere to the plant stack while introducing filtered makeup from the outside to provide adequate ventilation for personnel comfort when the plant is shut down during refueling operations and maintenance. It has two 100% capacity supply fans and two 100% capacity exhaust fans with filters and heating coil to temper the supply air and a filter train for the exhaust.

The low purge system is designed to continuously purge the containment during power operation to allow operator access. It has two 100% capacity supply fans and two 100% capacity exhaust fans with filters, heating coil, cooling coil to temper the supply air and a filter train for the exhaust.

The Containment Cooling and Ventilation System is not an Engineered Safety Feature and no credit has been taken for the operation of any subsystem or component in analyzing the consequences of design basis accidents.

Each Containment Purge Ventilation System supply and exhaust penetration through the containment vessel is equipped with two normally closed isolation valves, each connected to separate control trains. A failure in one train will not prevent the remaining isolation valve from providing the required isolation capability. The isolation valves and containment penetrations are the only portions of the Containment Purge Ventilation System that are engineered safety features.

Redundant containment isolation valves are designed, constructed, and tested in accordance with ASME Section III, Class 2. The valves are leak-tested periodically to verify acceptability of seat leakage. ~~Valves are designed to fail closed in the event of loss of power or loss of instrument air. All four Containment Purge isolation valves receive a containment isolation signal to close; however, the High Volume Purge System will not be open during power operation.~~

(continued)

BASES

BACKGROUND (continued)

~~The containment purge exhaust filter train is designed to meet the intent of Regulatory Guide 1.52. Ductwork from the containment penetration to the filter train will be low-leakage design.~~

~~The containment purge exhaust system is isolated on high radiation or high relative humidity signals. Relative humidity is controlled and monitored upstream of the containment purge exhaust filter trains.~~

~~Open high purge valves, or a failure of the low purge valves to close, following an accident which releases contamination to the containment atmosphere would cause a significant increase in the offsite radiation dose. This could result in exceeding the dose limits of Reference 1.~~

The OPERABILITY requirements for Containment Isolation Valves help ensure that containment leak tightness is maintained during and after an accident by minimizing potential leakage paths to the environment. Therefore, the OPERABILITY requirements provide assurance that containment leak rates assumed in the accident analysis will not be exceeded.

APPLICABLE SAFETY ANALYSIS

The Containment Isolation Valve LCO was derived from the requirements related to the control of offsite radiation doses resulting from major accidents. As delineated in 10 CFR 100, the determination of exclusion areas and low population zones surrounding a proposed site must consider a fission product release from the core with offsite release based upon the expected demonstrable leak rate from the containment. This LCO is intended to ensure the offsite dose limits are not exceeded (actual containment leak rate does not exceed the value assumed in the safety analysis). As part of the containment boundary, Containment Isolation Valve and Containment Purge Valve OPERABILITY is essential to containment integrity. Therefore, the safety analysis of any event requiring isolation of containment is applicable to this LCO.

The DBAs which result in a release of radioactive material within containment are a Loss Of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), a Main Feedwater Line Break (MFLB), or a Control Element Assembly (CEA) ejection accident. In the analysis for each of these accidents, it is assumed that Containment Isolation Valves are either closed or function to close within the required isolation time following event

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

initiation. This ensures that potential leakage paths to the environment through Containment Isolation Valves (and Containment Purge Valves) is minimized.

The acceptance criteria applied to accidental releases of radioactive material to the environment are given in terms of total radiation dose received by a member of the general public who remains at the exclusion area boundary for two hours following the onset of a postulated fission product release. The limits established in Reference 1 are a whole body dose of 25 Rem or a 300 Rem dose to the thyroid from iodine exposure, or both.

The accident analysis assumes that within [30 seconds] a CIAS, isolation of the containment is complete and leakage terminated, except for the design leak rate, La. The containment isolation total response time of [30 seconds] includes signal delay, diesel generator startup (for loss of offsite power), and Containment Isolation Valve stroke times.

The single failure criteria required to be imposed in the conduct of plant safety analysis was considered in the design of the Containment Purge Valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves on each line are provided with diverse power sources. This arrangement was designed to preclude common mode failures from disabling both valves on a purge line. The high purge valves may be unable to close in the environment following a LOCA. Therefore, each of the high purge valves is required to remain closed during MODES 1, 2, 3 and 4. In this case, the single failure criteria remains applicable to the containment purge valve arrangement; however, the concern is now the spurious opening of a purge valve due to failure in the control circuit associated with each valve. Again, the purge system valve design precludes a single failure from compromising containment integrity as long as the system is operated in accordance with the subject LCO.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The low purge valves are capable of closing under accident conditions. Therefore, they are allowed to be open for limited periods during power operation.

The Containment Isolation Valves and Containment Purge Valves satisfy the requirements of Selection Criteria 3 of the NRC Interim Policy Statement as documented in Reference 4.

LCO

Containment Isolation Valves form part of the containment boundary. The Containment Isolation Valve safety function is related to control of offsite radiation exposures resulting from a DBA. This LCO addresses Containment Isolation Valve structural integrity, stroke time and Containment Purge Valve leakage. The other Containment Isolation Valve leakage rates are addressed by LCO 3.6.1, Containment, under Type C testing.

The Containment Isolation Valves are considered OPERABLE when their isolation times are within limits and they isolate on an isolation actuation signal. The Containment Purge Valves have different OPERABILITY requirements. The [24-inch] purge valves must be maintained closed and all purge valves with resilient seals must meet special leak rate requirements.

The valves covered by this LCO are listed with their associated stroke times in the System 80+ CESSAR Design Certification (Ref. 2). This listing also indicates those valves may be opened/closed on an intermittent basis under administrative controls.

This LCO provides assurance that the Containment Isolation Valves and Purge Valves will perform their designed safety function to mitigate the consequences of accidents that could result in offsite exposure comparable to the Reference 3 limits.

APPLICABILITY

In MODES 1, 2, 3 and 4, a DBA could cause a release of radioactive material to containment. However, in MODES 5 and 6 the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the Containment Isolation Valves are not required to be OPERABLE and the Purge Valves are not required to be sealed closed in MODE 5. In MODE 6, fuel handling evolutions are

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BASES

APPLICABILITY (continued)

conducted. The requirements for Containment Isolation Valves and Containment Purge Valves during MODE 6 refueling operations are addressed in LCO 3.9.3, Containment Penetrations.

ACTIONS

A.1, A.2.1, A.2.2.1, and A.2.2.2

With one or more of the Containment Isolation Valves inoperable, except for purge valve leakage, at least one isolation valve must be verified to be OPERABLE in each affected open penetration. This action may be satisfied by examining logs or other information to determine if the valve is out of service for maintenance or other reasons. It does not mean to perform the SRs needed to demonstrate OPERABILITY of the valve. This Required Action is to be completed within one hour in order to provide assurance that a containment penetration is not open causing a loss of Containment integrity. The one-hour Completion Time is consistent with LCO 3.6.1, Containment, and is considered a reasonable length of time need to complete the Required Action.

In the event one or more Containment Isolation Valves are inoperable, except for purge valve leakage, either the inoperable valve must be restored to OPERABLE status or the affected penetration must be isolated. The method of isolation must include the use of at least one isolation barrier which can not be adversely affected by a single active failure. Isolation barriers which meet this criteria are a closed and deactivated automatic Containment Isolation Valve, a closed manual valve, a blind flange, or a check valve inside containment with flow through the valve secured. One of these two Required Actions must be completed within the four hour Completion Time. four-hour Completion Time is reasonable considering the time required isolate the penetration and the relative importance of maintaining containment integrity during MODES 1, 2, 3 and 4.

For affected penetrations which cannot be restored to OPERABLE status within the four hour Completion Time and have been isolated in accordance with Required Action A.2.2.1, the affected penetrations must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation

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ACTIONS (continued)

A.1, A.2.1, A.2.2.1, and A.2.2.2

position should an event occur. The Completion Time for this verification is once per 31 days for valves outside containment and prior to entering MODE 4 from MODE 5, but not more often than 92 days for valves inside containment. The Completion Time of 31 days for the valves outside containment is appropriate considering the valves are operated under administrative control and the low probability of their misalignment, the specified time period of prior to entering MODE 4 from MODE 5, but not more often than 92 days for valves inside containment is reasonable considering the relative inaccessibility of the valves.

The Required Actions of Condition A are modified by a note allowing Containment Isolation Valves, except [24-inch] purge valves, to be opened intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a valid Containment Isolation Signal is indicated. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, these valves may not be opened under administrative control.

Required Action A.1 is further modified by a Note stating that Action A.1 is not applicable to those penetrations with only one Containment Isolation Valve. Since the note to Condition A excludes penetrations with only one isolation valve and a closed system, the note to A.1 refers to penetrations with a single isolation valve on a system which is open inside containment. For these systems, if the single isolation valve is inoperable, the intent is to go directly to Action A.2. The four-hour Completion Time is reasonable for this situation because such systems typically have very small penetrations (e.g., sampling or instrumentation lines).

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ACTIONS (continued)

B.1, B.2.1 and B.2.2

With one or more Containment Isolation Valve(s) inoperable, the inoperable valve(s) must be restored to OPERABLE status or the affected penetration must be isolated. The method of isolation must include the use of at least one isolation barrier which can not be adversely affected by a single active failure. Isolation barriers which meet this criteria are a closed and deactivated automatic valve, a closed manual valve, or a blind flange. A check valve may not be used to isolate the affected penetration, since GDC 57 does not consider the check valve an acceptable automatic isolation valve. One of these Required Actions must be completed within the four hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary. In the event the affected penetration is isolated in accordance with Required Action B.2.1, the affected penetration must be verified to be isolated on a periodic basis. This is necessary to ensure containment integrity is maintained and that containment penetrations required to be isolated following an accident are isolated. Verification that each affected penetration is isolated once per 31 days is reasonable considering the importance of these valves and the low probability of their misalignment.

Condition B is modified by a note indicating this condition is only applicable to those penetrations with only one containment Isolation Valve and a closed system inside containment. This note is necessary since this condition is written to specifically address those penetrations isolated in accordance with 10 CFR 50 Appendix A, General Design Criteria (GDC) No. 57. GDC 57 allows lines that enter ACTIONS containment and which are not part of the reactor coolant pressure boundary nor connected directly to containment atmosphere, to be isolated by means of one Containment Isolation Valve.

C.1, C.2.1 and C.2.2

In the event one or more Containment Purge Valves ~~is~~ are not within the purge valve leakage limits, purge valve leakage must be restored to within limits or the affected penetration must be isolated. The method of isolation must be by the use of at least one isolation barrier which cannot be adversely affected by a single active failure. Isolation barrier which meet this criteria are a

(continued)

BASES

ACTIONS (continued)

C.1, C.2.1 and C.2.2

closed and deactivated automatic valve, closed manual valve, or blind flange. One of these Required Actions must be completed within the 24-hour Completion Time. The specified time period is reasonable considering the Containment Purge Valves remain closed such that a gross breach of containment integrity does not exist. For Containment Purge Valves which are isolated in accordance with Required Action C.2.1, SR 3.6.3.7 must be performed at least once per 92 days. This ensures that degradation of the resilient seals is detected and confirms that the leakage rate of the Containment Purge Valves does not increase during the time the penetration is isolated. The normal frequency of SR 3.6.3.7 is 184 days and is based on an NRC initiative contained in Generic Issue (GI B-20) "Containment Leakage Due to Seal Deterioration" (Ref. 5). Since somewhat more reliance is being placed on a single valve in this condition, it is prudent to perform the SR more often. Therefore, once per 92 days was chosen.

D.1 and D.2

The plant must be placed in a MODE in which the LCO does not apply if the inoperable Containment Isolation Valve cannot be restored or isolated, or Containment Purge Valve leakage can not be restored to within limits or isolated in the associated Completion Time. This is done by placing the plant in at least MODE 3 in six hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each [24-inch] Containment Purge Valve is required to be verified sealed closed at 31-day intervals. This surveillance ensures that a gross breach of containment is not caused by an inadvertent or spurious opening of a Containment Purge Valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to prevent offsite dose limits from exceeding 10 CFR 100 limits (Ref. 1). Therefore, these valves are required to be sealed closed position during MODES 1, 2, 3 and 4. A Containment Purge Valve that is closed must have

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.1

motive power to the valve operator removed. This can be accomplished by deenergizing the source of electric power or removing the air supply to the valve operator. The surveillance interval is a result of an NRC initiative (Generic Item B-24) related to Containment Purge Valve use during plant operations (Ref. 6).

SR 3.6.3.2

This SR ensures the [six-inch] purge valves are closed as required, or, if open, open for an allowable reason. This SR has been modified by a note indicating that these valves may be opened for pressure control, ALARA, and air quality considerations for personnel entry, and for surveillance tests that require the valve to be open. The [six-inch] purge valves are capable of closing under accident conditions. Therefore, these valves are allowed to be open for limited periods of time. The 31-day surveillance interval is consistent with other Containment Isolation Valve requirements discussed under SR 3.6.3.3.

SR 3.6.3.3

This SR verifies that all Containment Isolation manual valves and blind flanges which are located outside containment and required to be closed during accident conditions are closed. The SR helps to ensure that post-accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. The 31-day frequency is appropriate since these valves are operated under procedural control and the probability of an event requiring containment isolation during this time period is low. This frequency has been shown to be acceptable through operating experience.

Several notes have been added to this SR. The first note applies to valves and blind flanges located in high radiation areas, and allows these valves to be verified closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these valves once they have been verified to be in the proper position, is small. A second note has been

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.3

added which allows valves to be opened under administrative controls. These administrative controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way the penetration can be rapidly isolated when a valid CIAS ~~containment Isolation Signal~~ is indicated. A third note has been included to clarify that valves which are open under administrative controls are not required to meet the SR during the time the valves are open.

SR 3.6.3.4

This SR verifies that all containment isolation manual valves and blind flanges which are located inside containment and required to be closed during accident conditions are closed. The SR helps to ensure that post-accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For valves inside containment, the surveillance interval of prior to entering MODE 4 from MODE 5 but not more often than once per 92 days is reasonable considering the relative inaccessibility of these valves.

A note has been added to this SR which allows valves to be opened intermittently under administrative controls. The administrative controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way the penetration can be rapidly isolated when a valid containment Isolation Signal is indicated. An additional note has been included to clarify that valves which are open under administrative controls are not required to meet the SR during the time the valves are open.

SR 3.6.3.5

Demonstrating the isolation time of each power-operated and automatic Containment Isolation Valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The isolation time and frequency of this SR are in accordance with the Inservice Inspection and Testing Program.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.6

Automatic Containment Isolation Valves actuate on a Containment Isolation Signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures each automatic Containment Isolation Valve will actuate to its isolation position on a Containment Isolation Signal. The 18-month frequency was developed considering it is prudent that this SR only be performed during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for unplanned plant transients if the SR is performed with the reactor at power. Although the actuation logic is tested as part of the ESFAS functional test every 92 days, the subgroup relays that actuate the system cannot be tested during normal plant operation. Operating experience has shown that the components virtually always pass the SR when performed on the 18-month frequency which is consistent with the refueling cycle.

SR 3.6.3.7

For Containment Purge Valves with resilient seals, additional leak rate testing beyond the test requirements of 10 CFR 50, Appendix J is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. The 184-day frequency is the result of an NRC Staff initiative contained in Generic Issue (GI B-20) "Containment Leakage Due to Seal Deterioration" (Ref. 5). NUREG 1366 (Ref. 7) documents a recent reevaluation of the SR frequency that concluded the 184-day frequency is adequate. Additionally, this SR must be performed within 92 days after opening the valve. Ninety-two days was chosen recognizing that cycling the valve could introduce additional seal degradation (as opposed to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

A note has been added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1. This ensures that excessive Containment Purge Valve leakage will not cause the overall allowable containment leak rate to be exceeded and the containment rendered inoperable.

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BASES

REFERENCES

1. 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. System 80+ CESSAR-DC, Section 6.2, Containment Systems.

3. System 80+ CESSAR-DC, Chapter 15, Accident Analysis.

4. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director, NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application)."

5. Generic Issue (GI B-20), "Containment Leakage Due to Seal Deterioration."

6. NRC Generic Item B-24, "Purge Valve Reliability."

7. NUREG 1366, "Improvements to Technical Specification Surveillance Requirements."

Additional References

8. 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

9. 10 CFR 50, Appendix A, GDC 50, "Containment Design Basis."

10. 10 CFR 50, Appendix A, GDC 52, " Capability for Containment Leakage Rate Testing."

11. 10 CFR 50, Appendix A, GDC 53, "Provisions for Containment Inspection and Testing."

12. 10 CFR 50, Appendix A, GDC 54, "Piping Systems Penetrating Containment."

13. 10 CFR 50, Appendix A, GDC 56, "Primary Containment Isolation."

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BASES

REFERENCES (continued)

14. 10 CFR 50, Appendix A, GDC 57, "Closed System Isolation Valves."

15. BTP CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment."

16A.9.4 B 3.6.4 CONTAINMENT PRESSURE

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment structure serves to contain radioactive material which may be released from the reactor core following a Design Basis Accident (DBA), such that offsite radiation exposures are maintained within the requirement of 10 CFR 100 (Ref. 1). The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a Loss Of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in order to withstand an inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable which is monitored and controlled during MODES 1 through 4. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a DBA, a loss of containment integrity may result. Loss of containment integrity could cause site boundary doses, due to a DBA, to exceed values given in Reference 3.

APPLICABLE SAFETY ANALYSES

The limits for containment pressure ensure that operation is maintained within the design and accident analysis bases for containment. The accident analyses and evaluations considered both LOCAs and MSLBs for determining the maximum peak containment pressure (Pa) of [48.3] psig. A double-ended rupture of a main steam line at [0%] THERMAL POWER concurrent with a loss of one containment spray division results in the highest calculated internal containment pressure, [48.3] psig. This is below the internal design pressure of [~~49~~] 53 psig. The MSLB event bounds the LOCA event from the containment peak pressure standpoint.

The initial pressure condition used in t[illegible] analysis was 15.1 psia (0.4 psig). This resulted in a maxin[illegible]essure from a MSLB of [48.3] psig. The LCO limit of [0.4] psig ensures that in the event of an accident, the maximum accident design pressure for containment of [~~49~~] 53 psig

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BASES

APPLICABLE SAFETY ANALYSES (continued)	is not exceeded. The containment was also designed for an internal pressure equal to [-2.0] psid in order to withstand the resultant pressure drop from an accidental actuation of the Containment Spray System. ~~The initial pressure used in this analysis was [-0.4] psig.~~ The maximum calculated differential pressure which would occur as a result of an inadvertent actuation of the Containment Spray System is [1.83] psid, (starting with an initial pressure of [-0.4] p The LCO of [-0.4] psig ensures that operation within design limits of [-2.0] psid is maintained. Containment pressure satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement as documented in Reference 4.
LCO	During a DBA, with an initial containment pressure less than or equal to the upper LCO limit, the resultant peak containment accident pressure is maintained below the containment design pressure. The LCO lower pressure limit ensures the containment does not exceed the design negative differential pressure. As a result, containment integrity is ensured.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6 the probability and consequences of a DBA are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining these containment pressure limits is not required in MODES 5 or 6 to protect containment integrity.
ACTIONS	A.1 With containment pressure not within the limits of the LCO, containment pressure must be restored within one hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The specified time period is consistent with the ACTIONS of LCO 3.6.1, containment, which requires the containment be restored to OPERABLE status within one hour.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

The plant must be placed in a MODE in which the LCO does not apply if containment pressure cannot be restored in the required time period. This is done by placing the plant in at least MODE 3 in six hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power without challenging the plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying containment pressure is within limits ensures that operation remains within the limits assured in the containment analysis. The 12-hour Frequency of this SR was developed considering operating experience related to containment pressure variations and pressure instrument drift during the applicable MODES, and the low probability of a DBA occurring between surveillances. Furthermore, the 12-hour frequency is considered adequate in view of other indications in the control room to alert the operator of an abnormal containment pressure condition.

REFERENCES

1. 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. System 80+ CESSAR-DC, Section 6.2, Containment Systems.

3. System 80+ CESSAR-DC, Chapter 15, Accident Analysis.

4. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director, NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application)."

Additional References

5. 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

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BASES

REFERENCES (continued)

6. 10 CFR 50, Appendix A, GDC 50, "Containment Design Basis."

7. 10 CFR 50, Appendix A, GDC 52, "Capability for Containment Leakage Rate Testing."

8. 10 CFR 50, Appendix A, GDC 53, "Provisions for Containment Inspection and Testing."

9. 10 CFR 50, Appendix A, GDC 56, "Primary Containment Isolation."

16A.9.5 B 3.6.5 CONTAINMENT AIR TEMPERATURE

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material which may be released from the reactor core following a Design Basis Accident (DBA), such that offsite radiation exposures are maintained within the requirements of 10 CFR 100 (Ref. 1). The containment average air temperature is to be maintained within a lower and upper limit during normal operation to preserve the initial conditions assumed in the accident analyses for a Loss Of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). Containment air temperature is a process variable which is monitored and controlled during MODES 1 through 4. Temperature measurements from specified locations are combined to determine an average air temperature.

The containment average air temperature limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. Should operation occur outside this limit concurrent with a DBA, a loss of containment integrity or a violation of NRC LOCA acceptancecriteria may result. Loss of containment integrity could cause site boundary doses, due to a design basis MSLB, to exceed values given in Reference 3.

APPLICABLE SAFETY ANALYSIS

The limit for containment average air temperature ensures that operation is maintained within the DBA analysis assumptions for containment. The accident analyses and evaluations considered both LOCAs and MSLBs for determining the maximum peak containment pressures and temperatures. The worst case MSLB generates larger mass and energy releases than the worst case LOCA. Thus, the MSLB event bounds the LOCA event from the containment peak pressure and temperature standpoint. The initial pre-accident temperature inside containment was assumed to be [110°F] (Ref. 3).

The initial containment average air temperature condition of [110°F] resulted in a maximum vapor temperature in containment of [405.7°F]. The containment average temperature limit of [110°F] ensures that in the event of an accident, the maximum design temperature for containment of [290°F]

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

is not exceeded. The consequence of exceeding this design temperature may be the potential for degradation of the containment structure under accident loads.

Containment average air temperature satisfies the requirements of Selection Criterion of the NRC Interim Policy Statement as documented in Reference 4.

LCO

During a DBA, with an initial containment average temperature less than or equal to the LCO temperature limit, the resultant peak accident pressure and temperature is maintained below the containment design limits. As a result, the ability of containment to perform its function is assured.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment air temperature limit is not required in MODES 5 or 6.

ACTIONS

A.1

With containment average air temperature not within the limit of the LCO, containment average air temperature must be restored within the eight hour Completion Time. The Required Action must be taken to return operation to within the bounds of the containment analysis. The eight-hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems or to prepare the plant for an orderly shutdown.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

The plant must be placed in a MODE in which the LCO does not apply if the containment average air temperature cannot be restored to OPERABLE status in the associated Completion Time. This is done by placing the plant in at least MODE 3 in six hours and in MODE 5 in 36 hours. The allowable Completion Times are reasonable based on operating experience to reach the required MODES from full power without challenging the plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying the containment average air temperature is within the LCO limit ensures that containment operation remains within the limits assumed for the containment analysis. In order to determine the average temperature, an arithmetic average is calculated using measurements taken at several locations which are selected to be representative of the overall containment atmosphere. The 24 hour frequency of this surveillance is based on engineering judgment.

REFERENCES

1. 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. System 80+ CESSAR-DC, Section 6.2, Containment Systems.

3. System 80+ CESSAR-DC, Chapter 15, Accident Analysis.

4. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director, NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application)."

Additional References

5. 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal."

6. 10 CFR 50, Appendix A, GDC 39, "Inspection of Containment Heat Removal System."

(continued)

BASES

REFERENCES (continued)

7. 10 CFR 50, Appendix A, GDC 40, "Testing of Containment Heat Removal System."

8. 10 CFR 50, Appendix A, GDC 50, "Containment Design Basis."

9. 10 CFR 50, Appendix A, GDC 53, "Provisions for Containment Testing and Inspection."

16A.9.6 B 3.6.6 CONTAINMENT SPRAY SYSTEM

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray System (Credit taken for iodine removal by the Containment Spray System)

BASES

BACKGROUND

The Containment Spray System is designed to furnish containment atmosphere cooling to limit post-accident building temperature and pressure to less than the design values ([49] psig and [290°F], Ref. 1). Additionally, it reduces the release of radioactive material from the containment in the event of a primary or secondary break (the limiting events are a Loss Of Coolant Accident (LOCA) and a Main Steam Line Break (MSLB)) in two ways:

1. Reduction of containment pressure to nearly atmospheric pressure thereby reducing the potential leakage rate from containment; and

2. The boric acid solution minimizes the fission product iodine in the building atmosphere by the removal of iodine through the absorption of iodine by the spray droplets.

In the event of a LOCA or MSLB, the Containment Spray System sprays boric acid solution into the containment atmosphere to reduce the post-accident energy and to remove fission product iodine. There are two redundant Containment Spray divisions. Each division consists of one pump, one Containment Spray Heat Exchanger, one Containment Spray header and associated piping, valves, instrumentation and controls. The pumps and remotely operated valves may be operated from the control room.

A two out of four containment pressure high-high signal from the Engineered Safety Features Actuation System generates a Containment Spray Actuation Signal (CSAS) which initiates containment spray operation. Upon receipt of a CSAS, the Containment Spray Header isolation valve opens and the containment spray pump starts in each of the two redundant divisions. The pumps take suction initially from the In-Containment Refueling Water Storage Tank (IRWST) and discharge through the containment spray heat exchangers and the spray header isolation valves and to their respective spray nozzle headers, then into the containment atmosphere.

(continued)

BASES

BACKGROUND (continued)

The Containment Spray System is capable of removing sufficient decay heat from the containment atmosphere following a DBA accident to maintain containment pressure and temperature within design limits.

The Containment Spray System protects the integrity of the containment by limiting the temperature and pressure that could be expected following a DBA. Protection of adequate containment leaktightness prevents leakage of radioactive material from containment. Loss of adequate containment leaktightness could cause site boundary doses, due to a design bases LOCA, to exceed values given in Reference 3.

APPLICABLE SAFETY ANALYSIS

The accident analysis considers the worst case single active failure in the power supply which results in minimum containment cooling. The analysis and evaluation show that under this scenario, the highest peak containment pressure is [48.3] psig (experienced during a MSLB), actual temperature of the containment structure however, remained below the maximum design temperature of [290]°F. (See Bases B 3.6.4 - Containment Pressure, and B 3.6.5 - Containment Air Temperature, for a detailed discussion.) The limiting event is a MSLB initiated at 0% RTP. The analysis also assumes that one Containment Spray division is operating and an initial (pre-accident) condition of [110°F] and [0.40] psig for containment temperature and pressure respectively.

The effect of an inadvertent containment Spray actuation was also analyzed. The inadvertent containment Spray actuation results in a maximum negative containment pressure of [-1.83] psig. The design containment pressure is [-2.0] psig, hence the inadvertent actuation of the containment Spray system will not exceed containment design limits. Additional discussion is provided in Bases 3.6.4, Containment Pressure.

Containment Spray System satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement as documented in Reference 4.

(continued)

BASES

LCO

During a DBA, one division of Containment Spray is required to maintain containment peak pressure and temperature below design limits. To ensure these requirements are met, two Containment Spray divisions must be OPERABLE during normal operations. This ensures minimum cooling requirements are met if a DBA then occurs concurrently with a loss of offsite power.

APPLICABILITY

In MODES 1, 2, 3 and 4, a DBA could cause a release of radioactive material to containment and an increase in Containment pressure and temperature requiring the operation of the Containment Spray divisions. The probability and consequences of such an event in MODES 5 and 6 are reduced due to the pressure and temperature limitations in these MODES. The Containment Spray System is not required to be OPERABLE in MODES 5 and 6 to protect the integrity of containment.

ACTIONS

A.1

With one Containment Spray division inoperable, the inoperable Containment Spray division must be restored to OPERABLE status within 72 hours. In this condition, the remaining OPERABLE cooling division is adequate to perform the Containment Cooling and Iodine removal function. However, the overall reliability is reduced because a single failure in the OPERABLE divisions could result in no Containment cooling and no iodine removal capability. The 72-hour Completion Time is based on the iodine removal function and is consistent with other Engineered Safety Feature Systems' Completion Times for loss of one redundant division.

B.1 and B.2

The plant must be placed in a MODE in which the LCO does not apply in the event the Containment Spray division is not restored to OPERABLE status within the associated Completion Time. This is accomplished by placing the plant in at least MODE 3 within six hours, and in MODE 5 within 84 hours. The six hours allotted to reach MODE 3 is a reasonable time based on operating experience to shutdown the plant without challenging

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

plant systems. The extended interval to reach MODE 5 is reasonable when considering the driving force for a release of radioactive material from the RCS is reduced in MODE 3.

C.1

With two Containment Spray divisions inoperable, the plant is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the Containment Spray flowpath provides assurance that the proper flowpaths will exist for Containment Spray operation. This SR does not apply to valves which are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves which cannot be inadvertently misaligned, such as check valves. A valve which receives an actuation signal is allowed to be in a non-accident position provided the valve will automatically reposition within the proper stroke time. This SR does not require any valve testing or manipulation. Rather, it involves verification that those valves outside containment and capable of being mispositioned, are in the correct position. The 31-day frequency is appropriate because the valves are operated under procedural control. An improper lineup would only affect a single division, and the probability of an event requiring Containment Spray actuation during this time period is low. This frequency has been shown to be acceptable through operating experience.

SR 3.6.6.2

Demonstrating each Containment Spray Pump flow versus head meets design requirements ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.2

Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The frequency shall be in accordance with the Inservice Inspection and Testing Program.

SR 3.6.6.3 and SR 3.6.6.4

These SRs demonstrate each automatic Containment Spray Valve actuates to its required position on an actual or simulated Safety Injection Actuation Signal, and that each Containment Spray Pump starts on receipt of an actual or simulated Safety Injection Actuation Signal. The 18-month frequency was developed considering it is prudent that these surveillances only be performed during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for unplanned plant transients if the SR is performed with the reactor at power. Although the actuation logic is tested as part of the ESFAS functional tests every 92 days, and equipment performance is monitored as part of the Inservice Testing Program, the system actuation cannot be tested during normal plant operation. Operating experience has shown these components virtually always pass the SR when performed on the 18-month frequency which is consistent with the refueling cycle.

SR 3.6.6.5

With the Containment Spray inlet valves closed and the spray header drained of any solution, low pressure air, or smoke can be blown through test connections. Performance of this SR demonstrates that each spray nozzle is unobstructed provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive nature of the design of the nozzle, a ten-year test interval is considered adequate to detect degradation in performance and is consistent with the recommendations of NUREG-1366 (Ref. 5).

(continued)

BASES

REFERENCES

1. System 80+ CESSAR-DC, Section 6.2, Containment Systems.

2. System 80+ CESSAR-DC, Chapter 15, Accident Analysis.

3. 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

4. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director, NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application)."

5. NUREG-1366, "Improvements to Technical Specification Surveillance Requirements."

Additional References

6. 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal."

7. 10 CFR 50, Appendix A, GDC 39, "Inspection of Containment Heat Removal System."

8. 10 CFR 50, Appendix A, GDC 40, "Testing of Containment Heat Removal System."

9. 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup."

10. 10 CFR 50, Appendix A, GDC 42, "Inspection of Containment Atmosphere Cleanup Systems."

11. 10 CFR 50, Appendix A, GDC 43, "Testing of Containment Atmosphere Cleanup Systems."

12. 10 CFR 50, Appendix A, GDC 44, "Cooling Water."

13. 10 CFR 50, Appendix A, GDC 50, "Containment Design Basis."

16A.9.7 B 3.6.7 HYDROGEN ANALYZERS

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Hydrogen Analyzers

BASES

BACKGROUND

Hydrogen Analyzers are required to monitor the hydrogen concentration in the containment following a primary or secondary break, such as a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB) in containment. Hydrogen may accumulate within containment following a primary break as a result of a metal-steam reaction involving the zirconium fuel cladding and the reactor coolant, radiolytic decomposition of the post-accident emergency cooling solutions, corrosion of metals by solutions used for emergency cooling and containment spray, and hydrogen in the Reactor Coolant System (RCS) at the time of a primary break. During a secondary break such as MSLB, hydrogen production will only result from the corrosion of metals and paints in containment, ~~due to the solutions used in the Containment Spray System (CS).~~

The Hydrogen Analyzers are post-accident Type A Category 1 instruments. As such they are used to determine when to initiate hydrogen recombination operation following a primary or secondary break in containment.

Two independent Hydrogen Analyzers have been provided and each is powered from a separate vital AC power source. Each of the analyzers can monitor either of the supply lines from containment. Within 30 minutes after a LOCA, both hydrogen analyzers are manually activated to monitor hydrogen levels and to alert the operators in the control room if hydrogen concentration exceeds 3.5%. The analyzers when actuated will continuously monitor hydrogen concentration levels between [0 and 10%]. Both analyzers have the capability to interface with two areas that have been selected to provide a representative sample of the containment atmosphere following an accident.

The Hydrogen Analyzers measure the hydrogen concentration in containment so that required operator actions (e.g., actuate the Hydrogen Recombiners in accordance with emergency procedures) may be taken to prevent the hydrogen concentration from exceeding the flammability limit of 4.0 v/o. This eliminates the potential for a breach of containment due to a hydrogen-oxygen reaction.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS

To evaluate the potential for hydrogen accumulation in containment following a LOCA, hydrogen generation (as a function of time following the initiation of the accident) is calculated. Conservative assumptions recommended in Reference 1 are used to maximize the amount of hydrogen calculated. Assuming containment isolation, the concentration of hydrogen that would result as a function of time is calculated with and without credit taken for mitigating systems.

The calculations confirm that when mitigating systems are actuated, in accordance with the emergency procedures, the peak hydrogen concentration in containment is less than 4.0 v/o.

Hydrogen may accumulate within containment following a LOCA (or CEA ejection) as a result of:

1. A metal-steam reaction between the zirconium fuel rod cladding and the reactor coolant.

2. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump.

3. Hydrogen in the RCS at the time of the LOCA, i.e., hydrogen dissolved in the reactor coolant, and hydrogen gas in the pressurizer vapor space.

4. Corrosion of metals exposed to Containment Spray and Safety Injection System solutions.

In Reference 2, the NRC specified that the Hydrogen Analyzers meet Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 5), and that the LCO be retained in Technical Specifications.

LCO

Two Hydrogen Analyzers must be OPERABLE with power from two independent safety-related power supplies. This assures operation of at least one Hydrogen Analyzer in the event of a worst case single active failure. Operation of at least one Hydrogen Analyzer will provide the operator with information to enable action to be taken to prevent the containment post LOCA hydrogen concentration from exceeding the flammability limit.

(continued)

BASES

APPLICABILITY

In MODES 1 and 2, two Hydrogen Analyzers provide the operator with the capability to measure hydrogen concentration in containment assuming a worst case single active failure and allow, if required, action to be taken to control the hydrogen concentration within containment below its flammability limit of 4.0 v/o following a LOCA (Ref. 3). This ensures containment integrity and prevents damage to safety-related equipment and instrumentation located within containment.

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be significantly less than that calculated for the DBA LOCA. Thus, if the hydrogen analysis were to be performed starting with a LOCA in MODE 3 or 4, the time to reach a bulk concentration of 4.0 v/o would be extended beyond the conservatively calculated for MODES 1 and 2. The extended time would allow containment atmosphere sampling by other means to determine the hydrogen buildup, if the event the Hydrogen Analyzers are not available. Therefore, Hydrogen Analyzers are not required to be OPERABLE in MODES 3 and 4.

In MODES 5 and 6, the probability and consequences of a LOCA or MSLB are reduced due to the pressure and temperature limitations. The Hydrogen Analyzers are not required in these MODES to protect the integrity of containment.

ACTIONS

A.1

With one Hydrogen Analyzer inoperable, the inoperable analyzer must be restored to OPERABLE status within 30 days. The 30-day Completion Time is based on the low probability of the occurrence of a primary or secondary break that would generate hydrogen in amounts capable of exceeding the flammability limit, and the length of time after the event that operator action would be required to prevent exceeding this limit, and the availability of the Hydrogen Recombiners, the Hydrogen Purge System, and the Post-Accident Sampling System.

Required Action A.1 is modified by a note which indicates the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one Hydrogen Analyzer is inoperable. This allowance is provided because the probability of the occurrence of a primary or secondary break that would

(continued)

BASES

ACTIONS (continued)

A.1

generate hydrogen in amounts capable of exceeding the flammability limit is low, the probability of the failure of the OPERABLE analyzer is low, and the length of time after a postulated primary or secondary break before operator action would be required to prevent exceeding the flammability limit.

B.1

With two Hydrogen Analyzers inoperable, at least one analyzer must be restored to OPERABLE status within seven days. The seven-day Completion Time is based on the low probability of the occurrence of a primary or secondary break that would generate hydrogen in amounts capable of exceeding the flammability limit, and the length of time after the event that operator action would be required to prevent exceeding this limit, and the availability of the Hydrogen Recombiners, and the Post-Accident Sampling System.

C.1

The plant must be placed in a MODE in which the LCO does not apply if an inoperable Hydrogen Analyzer cannot be restored to OPERABLE status in the associated Completion ACTIONS Time. This is done by placing the plant in at least MODE 3 in six hours. The allowable Completion Times are reasonable based on operating experience to reach the required MODES from full power without challenging the plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire channel will perform its intended function.

The 92-day frequency has been shown to be acceptable through operating experience and is consistent with the recommendations of NUREG-1366 (Ref. 4).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.2

Performance of a CHANNEL CALIBRATION on the Hydrogen Analyzers using sample gases ensures the OPERABILITY of the analyzers is maintained. A typical CHANNEL CALIBRATION includes a minimum of two data points to verify accuracy of the analyzers over the range of interest. The sample gases used for performing the surveillances are nominally v/o hydrogen ≥ [0.98] and ≤ [1.02] (balance nitrogen), and nominally four v/o hydrogen ≥ [3.92] and ≤ [4.08] (balance nitrogen). The lower hydrogen flammability limit is assumed as 4.0 v/o hydrogen in air or steam-air atmospheres. Therefore, calibration with these sample gases helps ensure accurate information regarding containment hydrogen concentrations up to and including the flammability limit is available to operators following a LOCA. The 18 month frequency has been shown to be acceptable through operating experience and is consistent with the recommendations of NUREG-1366 (Ref. 4).

REFERENCES

1. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in containment Following a Loss-of- Coolant Accident, Revision 2, November 1978.

2. "NRC Staff Review of Nuclear Steam Supply Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," transmitted by Thomas E. Murley (NRC) letter to Joseph K. Gasper (CEOG) dated May 9, 1988.

3. System 80+ CESSAR-DC, Section 6.2, Containment Systems.

4. NUREG-1366, "Improvements to Technical Specification Surveillance Requirements."

5. 52 FR 3788, NRC Interim Policy Statement, on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

(continued)

BASES

REFERENCES (continued)

Additional References

6. 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup."

7. 10 CFR 50, Appendix A, GDC 42, "Inspection of Containment Atmosphere Cleanup Systems."

8. 10 CFR 50, Appendix A, GDC 43, "Testing of Containment Atmosphere Cleanup Systems."

REACTOR

16A.9.8 B 3.6.8 ~~CONTAINMENT~~ SHIELD BUILDING

B 3.6 CONTAINMENT SYSTEMS

Reactor
B 3.6.8 ~~Containment~~ Shield Building

BASES

BACKGROUND

Reactor S B
The ~~containment~~ shield building is a reinforced concrete structure composed of a right cylinder with a hemispherical dome. The ~~containment~~ Reactor S shield building houses the steel containment vessel and safety related equipment. The ~~containment~~ Reactor shield building is designed to provide biological shielding as well as external missile protection for the steel containment shell and safety related equipment. Between the steel containment vessel and the shield building inner wall is an annular space which collects any containment leakage which may occur following an in-containment break such as a Loss Of Coolant Accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

Following a LOCA, the Annulus Ventilation System (AVS) establishes a negative pressure in the annulus between the ~~containment~~ Reactor shield building and the Steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment. ~~Containment~~ Reactor Shield Building OPERABILITY is required to ensure retention of primary containment leakage and proper operation of the AVS.

APPLICABLE SAFETY ANALYSIS

The design basis for ~~containment~~ Reactor shield building OPERABILITY is a large break LOCA. Maintaining shield building integrity ensures that the release of radioactive materials from the primary containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analysis. This restriction, in conjunction with the operation of the AVS, will limit the site boundary radiation doses to within the limits of 10 CFR 100 (Ref. 3) during an accident.

LCO

~~Containment~~ Reactor shield building OPERABILITY must be maintained to ensure proper operation of the AVS and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analysis.

(continued)

BASES

APPLICABILITY

Maintaining of ~~containment~~ Reactor shield building OPERABILITY prevents leakage of radioactive material from the ~~containment~~ Reactor shield building. Radioactive material may enter the ~~containment~~ Reactor shield building from the primary containment following an in-containment break. Therefore, ~~containment~~ reactor shield building OPERABILITY is required during the same operating conditions which require primary containment OPERABILITY.

Primary containment OPERABILITY and ~~containment~~ reactor shield building OPERABILITY are required in MODES 1, 2, 3 and 4 when a Main Steam or Feed Line Break, LOCA or CEA ejection could release radioactive material to the primary containment atmosphere. In MODES 5 and 6 the probability and consequences of these events are low due to the reactor coolant system temperature and pressure limitations in these MODES. Therefore, neither containment OPERABILITY ~~or containment~~ Reactor shield building OPERABILITY is required in MODES 5 or 6.

ACTIONS

A.1

In the event ~~containment~~ reactor shield building OPERABILITY is not maintained, containment shield building OPERABILITY must be restored within 24 hours. The 24 hour completion time is based on engineering judgment.

B.1 and B.2

The plant must be placed in a MODE in which the requirement does not apply when ~~containment~~ reactor shield building OPERABILITY cannot be restored in the required time period. This is done by placing the plant in at least MODE 3 in six hours and in MODE 5 in 36 hours. The allowed completion time are reasonable based on operating experience to reach the required modes from full power without challenging safety systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

Maintaining shield building OPERABILITY requires maintaining each door in the access opening closed except when the access opening is ~~when~~ being used for normal ~~transient~~ transit entry and exit, ~~then at least one door must remain closed.~~ Stet Stet The surveillance frequency of 31 days is based on engineering judgment, and has been shown to be acceptable through operating experience.

SR 3.6.8.2

Maintaining the Reactor ~~Containment~~ Shield Building operable requires periodic inspection of the enclosure to detect any degradation. The surveillance frequency for this is the same as SR 3.6.1.1 and is done in conjunction with it.

REFERENCES

1. System 80+ CESSAR-DC, Section 3.8, Design of Category 1 Systems.
2. System 80+ CESSAR-DC, Section 6.2, containment Systems.
3. 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance."

16A.9.9 B 3.6.9 ANNULUS VENTILATION SYSTEM

3.6 CONTAINMENT SYSTEMS

3.6.9 Annulus Ventilation System

BASES

BACKGROUND

The Annulus Ventilation System (AVS) serves the space between the primary containment and the secondary containment. The system does not perform any normal ventilation function. However, it does provide additional assurance against the release of radioactivity to the environment; therefore, it is designed as an engineered safety feature and should be capable of operating and performing its function during startup, power operation, hot standby and hot shutdown.

Two independent ventilation divisions are provided. Each division consists of a fan, a filter train, associated ductwork, dampers, and controls as necessary to accomplish the design function. Each filter train consists of a moisture eliminator, ~~electrical heater~~, prefilter, and absolute filter, a carbon filter, and a post filter.

Each division also includes two annulus ventilation distribution ducts, one in the upper portion of the annulus and the other in the lower. Therefore, there are two ducts in the upper annulus and two in the lower, one for each division.

These distribution ducts contain grilles for annulus air intake and exhaust. The grilles of the upper distribution ducts draw air in from the annulus. This air passes through the moisture eliminator and the filter train before reaching the suction of the ventilation fan. The fan directs air either to the unit vent or both the unit vent and the lower annulus distribution duct. The grilles of the lower distribution ring expel air into the annulus.

See next page

The system will discharge sufficient air from the annulus to the unit vent to create a negative pressure of approximately [-0.5 in.] water gauge with respect to the outside atmosphere after a LOCA.

Two full capacity ventilation fans are provided with each one redundant of the other. The fans are supplied with power from the Class 1E Emergency Diesel Generators on LOOP.

(continued)

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

3. Annulus Ventilation System

The Annulus Ventilation System (AVS) serves the space between the primary containment and the secondary containment. The system does not perform any normal ventilation function. However, it does provide additional assurance against the release of radioactivity to the environment; therefore, it is designed as an engineering safety feature and should be capable of operating and performing its function during startup, power operation, hot standby and hot shutdown. The AVS has no effect on reactor criticality and is designed for public and station personnel radiation protection only.

System Function

The Annulus Ventilation System is designed to produce and maintain a negative pressure within the total annulus space and provides mixing of any in-leakage from the containment in order to preclude the unacceptable release of radioisotopes following a LOCA by providing fission product removal capability by decay and filtration.

System Configuration and Description

Two redundant ventilation divisions with independent active components are provided as shown in Figure III D-1. Each division consists of a fan, a filter train, associated ductwork, dampers, and controls as necessary to accomplish the design function. Each filter train consists of a moisture eliminator, an electrical heater, a prefilter, an absolute filter, a charcoal filter, and an absolute after filter.

Comments here have been incorporated in this copy

The two annulus ventilation divisions share one duct in the upper portion of the annulus and one duct in the lower portion of the annulus. Therefore, there is one common duct in the upper annulus and one common duct in the lower annulus for both systems. These distribution ducts contain grilles for annulus air intake and exhaust. The grilles of the upper duct draw air in from above the primary containment. This air passes through the moisture eliminator and the filter train before reaching the suction of the ventilation fan. The fan directs air either to the unit vent or both the unit vent and the lower annulus distribution duct. The grilles of the lower distribution ring expel air into the annulus.

The system will discharge sufficient air from the annulus to the unit vent to create a negative pressure of approximately -1 mm (-0.5 in) water gauge with respect to the outside atmosphere after a LOCA. The annulus ventilation distribution ducts permit the mixing of in-leakage in as large a volume as possible.

The design basis operation sequence is as follows:

- The Annulus Ventilation System is activated by the Containment Spray Actuation Signal.
- The fans exhaust air to the unit vent until annulus pressure is at negative pressure control point.

BASES

BACKGROUND (continued)

The moisture eliminator consists of a mechanical demister which is designed to remove entrained moisture droplets from the influent. An electric heater is provided to decrease the effluent relative humidity.

A are provided to adsorb

Each carbon filter ~~is sized to accommodate~~ the fission products released to the annulus following any of the postulated accidents. Failure of the carbon filter to perform the intended function will be detected by a unit vent radiation monitor, which monitors the activity level of the system effluent. The carbon filters are not credited for accident mitigation based on the source term calculations identified in Reference 15

APPLICABLE SAFETY ANALYSES

The purpose of this system is to produce and maintain a negative pressure zone in the annulus. This mitigates the consequences of airborne products of radiation that might otherwise become an environmental hazard during and following an accident.

The annulus ventilation system is designed and sized to meet the following criteria:

A. Produce and maintain a negative pressure within the total annulus space in order to preclude the unacceptable release of radioisotopes following an accident.

B. Provide fission product removal capability by decay and filtration.

C. Provide for the mixing of any in-leakage into the annulus space.

D. The design annulus in-leakage rate through the reactor shield wall and from the exterior atmosphere is [1000 SCFM at 0.5 in.] water differential pressure.

The system is designed to function during a seismic event; its location protects it from tornado/wind and missiles.

The system has no containment penetrations.

The system is 100% redundant which precludes single system failure.

(continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The system has complete electrical separation between the two divisions. Each division is powered by its respective Class 1E Emergency Diesel Generator.

The Annulus ventilation System is an engineered safety feature system.

The AVS satisfies the requirements of Selection Criteria 3 of the NRC Interim Policy Statement as documented in Reference 4.

LCOs

active components

Two independent and redundant divisions of the Annulus Ventilation System are required to be operable to ensure that at least one is available assuming a single failure disabling the other division.

APPLICABILITY

In MODES 1, 2, 3 and 4, a DBA could lead to fission product release to containment which leaks to the shield building. The large break LOCA, on which this system's design is based, is a full power event. Less severe LOCAs and leaks still require the system to be OPERABLE throughout these MODES.

The probability and severity of a LOCA decreases as core power and RCS pressure decrease. With the reactor shutdown the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequences of a DBA are very low due to the pressure and temperature limitations in these MODES. Under these conditions, the Annulus Ventilation system is not required to be OPERABLE.

ACTIONS

A.1

With one division of the Annulus Ventilation System inoperable, the inoperable division must be returned to OPERABLE status within seven days. The seven-day Completion Time is based on the low probability of a LOCA during this time period and the leaktightness of the containment (and is adequate to make most repairs).

(continued)

BASES

ACTIONS (continued)

B.1

With both divisions of AVS inoperable, action should be taken to restore one division to full OPERABLE status. The 24 hour Completion Time is based on engineering judgment, considering that any increase in releases because of system unavailability is low.

C.1 and C.2

The plant must be placed in a MODE in which the LCO does not apply if the inoperable AVS cannot be restored to OPERABLE status in the associated Completion Time. This is done by placing the plant in at least MODE 3 in six hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

A standby mode starts on demand and continues to operate. Standby systems should be checked periodically to ensure they start and function properly. As the environment and normal operating conditions on this system are not severe, testing each division once a month provides an adequate check on this system. Monthly heater operation drys out any moisture which may have accumulated in the charcoal from humidity in the ambient air.

SR 3.6.9.2

Testing the performance of the HEPA filter, ~~charcoal adsorber efficiency,~~ minimum system flow rate, and the physical properties of the activated carbon will be performed consistent with Regulatory Guide 1.52 requirements.

The 18-month frequency was developed considering it is prudent that these surveillances only be performed during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for unplanned plant transients if the SR is performed with the reactor at power. Although the actuation logic is tested as part of the ESFAS functional tests every 92

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.9.2

days, the system actuation cannot be tested during normal plant operation. Operating experience has shown these components virtually always pass the SR when performed on the 18-month frequency.

SR 3.6.9.3

This surveillance ensures that each AVS division responds properly. The 18-month frequency was developed considering it is prudent that these surveillances only be performed during a plant outage. This is due to the plant conditions needed to perform the SR and the potential for unplanned plant transients if the SR is performed with the reactor at power. Although the actuation logic is tested as part of the ESFAS functional tests every 92 days, the system actuation cannot be tested during normal plant operation. Operating experience has shown these components virtually always pass the SR when performed on the 18-month frequency which is consistent with the refueling cycle.

SR 3.6.9.4

[.25] in. gauge minimum

To limit leakage to the environs the annulus is kept at a slight negative pressure. This value is ~~must be~~ low enough to account for wind effects around the containment building (Ref. 10).

during accident conditions.

SR 3.6.9.5

To reduce the humidity of air entering the filter train, proper functioning of the heaters must be demonstrated periodically by measuring the power drawn by the heaters. It is impractical to test the system under conditions of high humidity, such as might be present from a leak spraying into the air, therefore the heaters are tested dry. A frequency of 18 months is specified in RG 1.52.

(continued)

BASES

REFERENCES

1. System 80+ CESSAR-DC, Section 6.2.3, Annulus Ventilation System.

2. System 80+ CESSAR-DC, Section 15.6.5, Loss of Coolant Accidents.

3. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application)."

Additional References

4. 10 CFR 50, Appendix A, GDC 41, "containment Atmosphere Cleanup."

5. 10 CFR 50, Appendix A, GDC 42, "Inspection of containment Atmosphere Cleanup Systems."

6. 10 CFR 50, Appendix A, GDC 42, "Testing of containment Atmosphere Cleanup Systems."

7. 10 CFR 50, Appendix A, GDC 61, "Fuel Storage and Handling and Radioactivity Control."

8. 10 CFR 50, Appendix A, GDC 64, "Monitoring of Radioactivity Releases."

9. Regulatory Guide 1.52 (Rev. 02), "Design, Testing and Maintenance Criteria for Post Accident Engineered Safety Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."

10. ANSI/ASME N509-1980, "Nuclear Power Plant Air Cleaning Units and Components." (NOTE: SRP 6.5.1, Rev. 2, July 1981 approved the use of the 1980 version, rather than the 1976 version in Reference 5).

(continued)

BASES

REFERENCES (continued)

11. ANSI/ASME N510-1980, "Testing of Nuclear Air Cleaning Systems." (NOTE: SRP 6.5.1, Rev. 2, July 1981 approved the use of the 1980 version, rather than the 1975 version in Reference 8).

12. NRC Generic Letter 83-13, "Clarification of Surveillance Requirements for HEPA Filters and Charcoal Adsorber Units in Standard Technical Specifications on ESF Cleanup Systems", March 2, 1983.

13. NUREG-0800, "Standard Review Plan", Section 15.6.5, Rev. 2, Appendix B, "Radiological Consequences of a Design Basis Loss of Coolant Accident: Leakage from Engineered Safety Feature Components Outside Containment," Rev. 1.

14. BTP CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment."

15. CESSAR-DC, Section 2.3 "Meteorology"

16A.10 B 3.7 PLANT SYSTEMS

16A.10.1 B 3.7.1 MAIN STEAM SAFETY VALVES

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves

BASES

BACKGROUND

The Main Steam Safety Valves (MSSVs) mainly provide over-pressure protection for the secondary system. In doing so, the MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary by providing a heat sink for removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water system, is not available.

Five Main Steam Safety Valves, (ten per steam generator) are located on each Main Steam Line, outside Containment, upstream of the Main Steam Isolation Valves, as described in CESSAR-DC Section 5.2.2 (Ref. 1). The MSSVs' rated capacity passes the full steam flow at 102% RATED THERMAL POWER (RTP)(100 + 2% for instrument error) with the valves full open. This meets the requirements of the ASME Code (Ref. 2) as described in the Over-pressure Protection Report, CESSAR-DC Appendix 5.A (Ref. 3). The MSSV design includes staggered setpoints, as shown in Table 3.7.1-2, so that only the number of valves needed will actuate. Staggered setpoints reduce the potential for valve chattering because of insufficient steam pressure to fully open all valves following a turbine-reactor trip.

The valve lift settings given in Table 3.7.1-2 meet the requirements of Section III of the ASME Code (Ref. 2). The total relieving capacity for all twenty MSSVs at 110% of system design pressure (adjusted for a 50 psi pressure drop to valves inlet) is 19 E6 lbm/hr plus accumulation. This capacity is less than the total rated capacity because the MSSVs operate at an inlet pressure below rated conditions ensuring that steam generator pressure does not exceed 110% of design. At these same secondary pressure conditions, the total steam flow at 102% of [3,817 Mwt] (RTP plus 17 Mwt pump heat input) is [17.46 E6 lbm/hr.] The ratio of this total steam flow to the total capacity is 109.9%.

The low pressure setpoint MSSV, 1200 psia, corresponds to a zero power loop average temperature (T_{ave}) (secondary fluid saturation temperature) of 566°F. The RCS T_{ave} must be above this temperature to open MSSVs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from the ASME Code and limits secondary system pressure to ≤ 110% of design pressure when passing 100% of design steam flow. This design basis is more than sufficient to cope with any anticipated operating occurrence (AOO) or accident considered in the Design Basis Accident and Transient Analysis. For most analyzed events, RCS pressure remains below the setpoint of the pressurizer safety valves (PSVs), or, at most, cause only a short opening of the PSVs.

The events that challenge the MSSVs' relieving capacity, and thus RCS pressure, are those characterized as Decreased Heat Removal events, and are presented in Section 15.2 of the CESSAR-DC (Ref. 4). Of these, the full power loss of condenser vacuum (LOCV) event is the limiting AOO. A LOCV isolates the turbine and condenser, and terminates normal feedwater flow to the Steam Generators. Before delivery of Emergency Feedwater (EFW) to the Steam Generators, RCS pressure reaches ≤ [2,630] psia. This peak pressure is less than 110% of the design pressure of 2,500 psia, but high enough to actuate the PSVs. The maximum relieving rate during the LOCV event is 2.5 E6 lbm/hour which is less than the rated capacity of three of the MSSVs.

The limiting accident for peak RCS pressure is the full power feedwater line break, inside Containment, with the failure of the backflow check valve in the feedwater line from the affected Steam Generator. Water from the affected steam generator is assumed to be lost through the break with minimal additional heat transfer from the RCS. With heat removal limited to the unaffected Steam Generator, the reduced heat transfer causes an increase in RCS temperature and the resulting RCS fluid expansion causes an increase in pressure. The RCS pressure increases to ≤ 2,730 psia, with the PSVs providing relief capacity. The maximum relieving rate of the MSSVs during the feedwater line break event is ≤ 2.5 E6 lbm/hour which is less than the rated capacity of two of the MSSVs.

Using conservative analysis assumptions, a small range of feedwater line break sizes, less than a full double-ended guillotine break, produces an RCS pressure of 2,765 psig for a period of 20 seconds; exceeding 110% (2,750 psia) of design pressure. This is considered acceptable as RCS pressure is still well below 120% of design pressure where deformation might happen. The probability of this event is also very low, in the range of 4 E-6/year.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening and failure to reclose once opened. The passive failure mode is failure to open upon demand.

The MSSVs satisfies the requirements of Criterion 3 of the Interim Policy Statement as described in Reference 5.

LCO

Three MSSVs per Steam Generator are required by the accident analysis to provide overpressure protection for design basis transients occurring at 102% of RTP. It is not necessary to assume a passive failure in the short term following an event. The LCO requires all MSSVs to be OPERABLE in compliance with the ASME Code, even though this is not a requirement of the design basis accident analysis. This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements) and adjustment to the Reactor Protective System Trip Setpoints. These limitations are addressed in Table 3.7.1-1 and RAs A.2.2.1 and A.2.2.2.

The operability of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator over-pressure and to re-seat when pressure has been reduced. The operability of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Inspection and Testing Program.

The lift settings specified in Table 3.7.1-2 correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety function to mitigate the consequences of accidents that could result in a challenge to the reactor coolant pressure boundary.

APPLICABILITY

In MODE 1, the accident analysis requires a minimum of three MSSVs per Steam Generator which is limiting and bounds all lower MODES. In MODES 2 and 3 both the ASME Code and the accident analysis only require one MSSV per Steam Generator to provide overpressure protection.

(continued)

BASES

APPLICABILITY (continued)

In MODES 4 and 5 there is no credible transient requiring the MSSVs.

In MODE 6, the Steam Generators are not used for heat removal and cannot be overpressurized.

ACTIONS

A.1

All MSSVs are required to be OPERABLE for each Steam Generator to meet ASME Code requirements at RTP. Three MSSVs for each Steam Generator have sufficient capacity to provide overpressure protection for design basis transients occurring at 102% of RTP. This Action may be satisfied by examining logs or other information to determine if the MSSVs are out-of service for maintenance or other reasons. It does not mean to perform the SRs needed to demonstrate OPERABILITY of the MSSVs. The four hour Completion Time is the same as that for restoring a MSSV to OPERABLE status, and is reasonable considering that only two MSSVs are required by the accident analysis.

A.2.1

With one or more MSSVs inoperable, one alternative is to restore the required inoperable equipment per Table 3.7.1-1. The four hour Completion Time to restore a MSSV to OPERABLE status is reasonable considering that only three MSSVs are required by the accident analysis.

A.2.2.1 and A.2.2.2

An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets Code requirements for the power level. Operation may continue provided the allowable THERMAL POWER is equal to the product of: 1) the ratio of the number of MSSVs available per Steam Generator to the total number of MSSVs per Steam Generator, and: 2) the ratio of the total steam flow to available relieving capacity multiplied by 100%.

$$\text{Allowable THERMAL POWER} = \left(\frac{10 - N}{10}\right) \times 109.9$$

(continued)

BASES

ACTIONS (continued)

The ceiling on the variable overpower trip is also reduced to an amount over the allowable THERMAL POWER equal to the Band given for this trip in Table 3.7.1-1.

SP = Allowable THERMAL POWER + 9.8

where:

SP	=	reduced reactor trip setpoint in percent RTP. This is a ratio of the available relieving capacity over the total steam flow at rated power.
10	=	total number of MSSVs per Steam Generator.
N	=	number of inoperable MSSVs on the Steam Generator with the greatest number of inoperable valves.
109.9	=	ratio of MSSV relieving capacity at 110% Steam Generator design pressure to calculated steam flow rate at 100% RTP + 2% instrument uncertainty expressed as a percentage (see text above).
9.8	=	Band between the maximum THERMAL POWER and the variable overpower trip setpoint ceiling (Table 3.7.1-1).

Limit the maximum steady state power level to some value slightly below this setpoint to avoid an inadvertent overpower trip.

The four-hour Completion Time for RA A.2.2.1 is consistent with A.1 and A.2.1. An additional four hours is allowed to reduce the setpoints in recognition of the difficulty of resetting all channels of this trip function within a period of four hours. The Completion Time of eight hours is based on operating unit experience in resetting all channels of a protective function and on the improbability of the occurrence of a transient which could result in Steam Generator overpressure. Completion of RA A.1 assures that the minimum number of MSSVs required per the accident analysis are OPERABLE.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

The plant must be placed in a MODE in which the requirement does not apply if the MSSVs cannot be restored to OPERABLE status in the associated Completion Time. This is done by placing the plant in at least MODE 3 in six hours and in MODE 4 in 12 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power operation without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR demonstrates the OPERABILITY of the MSSVs. Section XI, Article 3500 of the ASME Code (Ref. 6) Requires that safety and relief valve tests be performed as required by ASME/ANSI OM-1-1987 (Ref. 7). Section 7.3.2.1 of this standard requires the following tests for MSSVs:

1. Visual examination,
2. Seat tightness determination,
3. Set pressure determination,
4. Compliance with owner's seat tightness criteria, and
5. Verification of the balancing device integrity device on balanced valves.

The standard requires testing all valves every five years, with a minimum of 20% of the valves tested every 24 months. Surveillance requirements are specified in the Inservice Testing Program which encompasses Section XI of the ASME Code. ASME Code provides the activities and frequencies necessary to satisfy the requirements.

SR 3.7.1.1 is modified by a Note which allows an exemption to SR 3.0.4. The MSSVs may be either bench tested, or tested in-situ at hot conditions using an assist device to simulate lift pressure. The SR 3.0.4 exemption applies to those plants which have provisions for testing the MSSVs at hot conditions. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

(continued)

BASES

REFERENCES

1. CESSAR-DC Section 5.2, "Overpressure Protection."

2. ASME Boiler and Pressure Vessel Code, Section III, Article NC-7000, "Overpressure Protection" Class 2 Components.

3. CESSAR-DC Chapter 5, Appendix 5A, "Overpressure Protection Report."

4. CESSAR-DC Section 15.2, "Accident Analysis - Decreased Heat Removal Events."

5. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Standard Technical Specifications - Volume 1 (Criteria Application)."

6. ASME Boiler and Pressure Vessel Code, Section XI, Article IWV-3500, "Inservice Tests - Category C Valves."

7. ANSI/ASME OM-1-1987, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices.

Additional References

8. 10 CFR 50, Appendix A, GDC 15 - Reactor Coolant System Design.

9. NUREG-0800, "Standard Review Plan", Section 10.3 "Main Steam System."

16A.10.2 B 3.7.2 MAIN STEAM ISOLATION VALVES

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves

BASES

BACKGROUND

The Main Steam Isolation Valves (MSIVs) isolate steam flow from the secondary side of the Steam Generators following a high energy line break. MSIV closure terminates flow from the unaffected (intact) Steam Generator.

One MSIV is located in each Main Steam Line outside, but close to, Containment. The MSIVs are downstream from the MSSVs, ADVs and Emergency Feedwater Pump turbine's steam supplies to prevent their being isolated from the Steam Generators by MSIV closure. Two of the MSIVs (one per Steam Generator) have bypass valves that allow the warming of the downstream main steam piping. Closing the MSIVs isolates each Steam Generator from the other, and isolates the turbine, steam bypass system, and other auxiliary steam supplies from the Steam Generators.

The MSIVs and MSIV Bypass Valves close on a Main Steam Isolation Signal (MSIS) generated by either low steam generator pressure or high Containment pressure. The MSIVs and bypass valves fail close on loss of control or activation power. The MSIS also actuate the Main Feedwater Isolation Valves to close. The MSIVs and bypass valves may also be closed manually.

A description of the MSIVs and bypass valves is found in Section 10.3 of the CESSAR-DC (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the Containment analysis for the large steam line break inside Containment (Ref. 2). It is also influenced by the accident analysis of the steam line break events presented in Chapter 15.1 of the CESSAR-DC (Ref. 3). The design precludes the blowdown of more than one Steam Generator, assuming a single active component failure, i.e., the failure of one MSIV to close on demand.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting case for the Containment analysis is the hot zero power steam line break inside Containment with a loss of offsite power following turbine trip and failure of the MSIV on the affected Steam Generator to close. At zero power the Steam Generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the Containment. Failure of the MSIV to close contributes the additional mass and energy in the steam headers downstream of the other MSIV to the total releases via backflow. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the borated water injection delivered by the Emergency Core Cooling System. Other failures considered are the failure of a Main Feedwater Isolation Valve to close, and failure of an emergency diesel generator to start.

The accident analysis compares several different steam line break events against different acceptance criteria. The large steam line break outside Containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large steam line break inside Containment at hot zero power is the limiting case for a post-trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the Steam Generators, maximizing the RCS cooldown. With a loss of offsite power, the response of mitigating systems, such as the safety injection (SI) pumps is delayed. Significant single failures considered include: failure of an MSIV to close, failure of an emergency diesel generator, and failure of a SI pump.

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

1. High energy line break inside Containment. For this scenario, steam is discharged into Containment from both Steam Generators. Mass and energy release from a break results in pressure and temperature increases in Containment. Closure of the MSIVs isolates the break and limits the blowdown to a single Steam Generator.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2. A break outside of Containment and upstream from the MSIVs is not a Containment pressurization concern. The uncontrolled blowdown of more than one Steam Generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single Steam Generator.

3. A break downstream of the MSIVs will be isolated by the closure of the MSIVs. Events such as increased steam flow through the Turbine or the Steam Bypass valves will also terminate on closing the MSIVs.

4. Following a Steam Generator tube rupture, closure of the MSIVs isolates the affected Steam Generator from the intact Steam Generator. In addition to minimizing radiological releases, this enables the operator to establish a pressure difference between the ruptured and intact Steam Generators, a necessary step toward terminating the flow through the rupture.

5. The MSIVs are also utilized during other events such as a feedwater line break.

The MSIVs satisfy the requirements of Criterion 3 of the Interim Policy Statement as described in Reference 4.

LCO

This LCO requires that the MSIV in each of the steam lines be OPERABLE. The MSIVs are considered OPERABLE when their isolation times are within limits, and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents which could result in offsite exposures comparable to the 10 CFR 100 (Ref. 5) limits.

(continued)

BASES

APPLICABILITY

The MSIVs must be OPERABLE whenever there is significant mass and energy in the RCS and Steam Generators. In MODES 1, 2, and 3 there is significant mass and energy in the RCS and Steam Generators, therefore, the MSIVs must be OPERABLE or closed. When the MSIVs are closed they are already performing their safety function. This ensures that in the event of a high energy line break, a single failure can not result in the blowdown of more than one Steam Generator.

In MODE 4 the MSIVs are normally shut, and the Steam Generator energy is low.

In MODES 5 and 6, the Steam Generators do not contain much energy because their temperature is below the boiling point of water. Therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable, time is allowed to restore the component to OPERABLE status. Some repairs can be made to the MSIV with the plant hot. The eight hour Completion Time is reasonable considering the probability of an accident occurring during the time period which would require closure of the MSIVs.

The eight hour Completion Time is greater than that normally allowed for Containment isolation valves because the MSIVs are GDC-57 (Ref. 6) valves that isolate a closed system penetrating Containment. These valves differ from other Containment isolation valves in that the closed system provides additional support for the Containment isolation function.

B.1 and B.2

If the MSIV cannot be restored to OPERABLE status within eight hours, the MSIV must be closed within the next six hours. Six hours is a reasonable time to complete the actions required to close the MSIV, which includes performing a controlled plant shutdown to MODE 2. The Completion Time is based on plant operating experience related to the time required to reach MODE 2 with the MSIVs closed without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

B.3

If the MSIV Bypass Valve cannot be restored to operable status, then it must be placed in its required post-MSIS position. By closing the valve, it ensures proper isolation of the Main Steam system will occur in the event of an MSIS. Changing mode from MODE 1 to 2 is not required since the valve is normally closed during power operation.

C.1 and C.2

The plant must be placed in a MODE in which the requirement does not apply if the MSIVs cannot be restored to OPERABLE status or closed in the associated Completion Time. This is done by placing the plant in at least MODE 3 in six hours and in MODE 4 in 12 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power operation without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

The MSIV closure time is assumed in the accident and Containment analyses. This SR is normally performed upon returning the plant to operation following a refueling outage. The MSIVs are not testable at power. Even a part-stroke exercise increases the risk of a valve closure with the plant generating power. As the MSIVs cannot be tested at power, they are exempt from the ASME Section XI (Ref. 7) requirements during operation in MODES 1 and 2.

The MSIVs cannot be fully stroked as part of the SIAS CHANNEL FUNCTIONAL TEST during normal operation. This SR ensures that it is fully tested at least once per refueling cycle. The actuation logic is tested as part of the SIAS functional test every 92 days. The subgroup relays that actuate the system cannot be tested during normal plant operation. The surveillance interval of 18 months is based on the refueling cycle and has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.1 is modified by a Note which allows exemption to SR 3.0.4. SR 3.0.4 is not applicable to this SR for entry into MODE 3 for the purposes of testing the MSIVs. This allows delaying testing to MODE 3 in order to have conditions consistent with those under which the valves will be operated.

REFERENCES

1. CESSAR-DC Section 10.3, "Main Steam System."

2. CESSAR-DC Section 6.2, "Containment Analysis."

3. CESSAR-DC Section 15.1.5, "Steam Line Break Analysis."

4. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Standard Technical Specifications - Volume 1 (Criteria Application)."

5. 10 CFR 100, Reactor Site Criteria.

6. 10 CFR 50, Appendix A, GDC 57 - Closed System Isolation Valves.

7. American Society for Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400 "Inservice Tests - Category A and B Valves."

Additional References

8. 10 CFR 50, Appendix A, GDC 38 - Containment Heat Removal.

9. NRC Information Notice 88-51, "Failures of Main Steam Isolation Valves," July 21, 1988.

10. NUREG-0800, "Standard Review Plan", Section 10.3 "Main Steam Supply System," Rev. 3, April 1984.

11. NRC Regulatory Guide 1.148, "Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants."

16A.10.3 B 3.7.3 MAIN FEEDWATER ISOLATION VALVES

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves

BASES

BACKGROUND

The Main Feedwater Isolation Valves (MFIVs) isolate main feedwater flow to the secondary side of the Steam Generators following a high energy line break. Closure of the MFIVs terminates flow to both Steam Generators, terminating the event for feedwater line breaks occurring upstream of the MFIVs. The consequences of events occurring in the main steam lines or in the main feedwater lines downstream of the MFIVs will be mitigated by their closure. Closure of the MFIVs effectively terminates the addition of feedwater to an affected Steam Generator, limiting the mass and energy release to Containment for inside Containment steam or feedwater line breaks, and reducing the cooldown effects for steam line breaks.

The MFIVs isolate the non-safety related portions from the safety related portion of the system. In the event of a secondary side pipe rupture inside Containment, the valves limit the quantity of high energy fluid that enters Containment through the break and provides a pressure boundary for the controlled addition of Emergency Feedwater (EFW) to the intact Steam Generator.

Two MFIVs are located in series on each main feedwater downcomer line and each main feedwater economizer line (4 per steam generator), outside, but close to Containment. The valves in series offer redundant isolation of main feedwater to each Steam Generator. The MFIVs are located upstream of the Emergency Feedwater (EFW) injection point so that EFW may be supplied to an unaffected Steam Generator following MFIV closure. The large length of piping from the MFIVs to the Steam Generators, must be accounted for in calculating mass and energy releases, and must be refilled prior to EFW reaching the Steam Generator following either a steam or feedwater line break.

The MFIVs close on receipt of a Main Steam Isolation Signal (MSIS) generated by either low steam generator pressure or high Containment pressure. The MSIS also actuates the main steam isolation valves to close. The MFIVs may also be actuated manually.

(continued)

BASES

BACKGROUND (continued)

A description of the MFIVs is found in Section 10.4.7 of the CESSAR-DC (Ref. 1). The MFIVs and a check valve inside Containment are required to isolate the feedwater line penetrating Containment, and to ensure the consequences of events do not exceed the capacity of the Containment heat removal systems.

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs is established by the Containment analysis for the large steam line break inside Containment. It is also influenced by the accident analysis for the large feedwater line break, although the limiting RCS pressure occurs before MFIV closure. Closure of the MFIVs may also be relied on to terminate an excess feed event upon the generation of a MSIS on high steam generator level.

Failure of a MFIV to close contributes additional mass and energy to the Steam Generators in the case of a steam line break inside Containment. However, in order for the Main Feedwater (MFW) system to continue supplying feedwater, offsite power must be available. In such a case, the accident analysis assumes both trains of Containment spray/cooling are available (the MFIV being the assumed single failure) and are sufficient to mitigate the Containment pressure transient.

Although not credited in the accident analysis of such an event, several other backups are available to terminate MFW flow should a MFIV fail to close on receipt of an MSIS. Loss of offsite power stops any motor driven pumps, such as the condensate and MFW pumps. Even with power available, the MFW pumps would probably trip on low suction pressure or high vibrations due to the reduction in backpressure and the feedwater flashing to steam. The loss of condenser vacuum, upon closing the MSIVs, trips the MFW pumps, or condenser inventory runs dry after about 10 minutes. A turbine trip actuates the MFW control valves to throttle or close, although they may not be designed to close against accident differential pressures.

The MFIVs may also be actuated during a LOCA by MSIS on high Containment pressure. The MFIVs show little effect on the LOCA Containment analysis.

The MFIVs satisfies the requirements of Criterion 3 of the Interim Policy Statement as described in Reference 2.

(continued)

BASES

LCO

Following a feedwater or main steam line break, this LCO ensures that the MFIVs will isolate MFW flow to the Steam Generators. The MFIVs are considered OPERABLE when their isolation times are within limits, and they close on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to Containment following a steam or feed line break inside Containment. If MSIS on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.

This LCO provides assurance that the MFIVs will perform their design safety function to mitigate the consequences of accidents which could result in challenging the integrity of Containment, the reactor coolant pressure boundary or the secondary piping.

APPLICABILITY

In MODES 1, 2, and 3 the MFIVs are required to be OPERABLE to limit the amount of fluid available to be added to the Containment in the case of a secondary system pipe break inside Containment.

In MODES 4, 5, and 6 main feedwater is not required and the MFIVs are normally closed.

ACTIONS

A.1 and A.2

With one MFIV inoperable, the Required Action is to restore the valve to OPERABLE status or to close or isolate the valve. The plant must be placed in MODES 2 or 3 with the MFIV closed if the valve is not restored to OPERABLE status. The 72 hour Completion Time to restore or close the valve is acceptable due to the low probability of events requiring the MFIVs and the availability of redundant valves to isolate main feedwater flow.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

With both the MFIV's in the same flow path inoperable in one or more flow paths, the Required Action is to restore at least one valve in the affected flow paths to OPERABLE status or to close or isolate at least one valve. The plant must be placed in MODE 3 with the MFIV closed if at least one valve is not restored to OPERABLE status.

The MFIVs are GDC-57 (Ref. 3) valves that isolate a closed system that penetrates Containment. These valves differ from other Containment isolation valves in that the closed system provides additional support for the Containment isolation function. The eight hour Completion Time to restore or close at least one valve is comparable to that for MSIVs. This is acceptable due to the low probability of the events requiring the MFIVs and the availability of backups by non-safety grade features to terminate main feedwater flow.

C.1 and C.2

The plant must be placed in a MODE in which the requirement does not apply if the MFIVs cannot be restored to OPERABLE status or closed in the associated Completion Time. This is done by placing the plant in at least MODE 3 in six hours and in MODE 4 in 12 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power operation without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

The MFIV closure time is assumed in the accident and Containment analyses. This SR is normally performed upon returning the plant to operation following a refueling outage. As the MFIVs cannot be tested at power, they are exempt from the ASME Section XI (Ref. 4) requirements during operation in MODES 1 and 2.

(continued)

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BASES

SURVEILLANCE REQUIREMENTS (continued)

The MFIVs cannot be fully stroked as part of the SIAS CHANNEL FUNCTIONAL TEST during normal operation. This SR ensures that the MFIVs are fully tested at least once per refueling cycle. The actuation logic is tested as part of the SIAS functional test every 92 days. The subgroup relays that actuate the system cannot be testing during normal plant operation. The surveillance interval of 18 months is based on the refueling cycle and has been shown to be acceptable through operating experience.

SR 3.7.2.1 is modified by a Note which allows exemption to SR 3.0.4. SR 3.0.4 is not applicable to this SR for entry into MODE 3 for the purposes of testing the MFIVs. This allows delaying testing to MODE 3 in order to have conditions consistent with those under which the valves will be operated.

REFERENCES

1. CESSAR-DC Section 10.4.7, Condensate and Feedwater System.

2. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Standard Technical Specifications - Volume 1 (Criteria Application)."

3. 10 CFR 50, Appendix A, GDC 57 - Closed System Isolation Valves.

4. American Society for Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400 "Inservice Tests - Category A and B Valves."

Additional References

5. CESSAR-DC Section 6.2.1.4, Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures Inside Containment.

6. CESSAR-DC Section 15.2.5, Feedwater Line Breaks.

7. 10 CFR 50, Appendix A, GDC 38 - Containment Heat Removal.

(continued)

BASES

REFERENCES (continued)

8. Branch Technical Position ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to NUREG-0800, "Standard Review Plan", Section 3.6.1.

9. Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to NUREG-0800, "Standard Review Plan", Section 3.6.2.

16A.10.4 B 3.7.4 EMERGENCY FEEDWATER SYSTEM

B 3.7 PLANT SYSTEMS

B 3.7.4 Emergency Feedwater System

BASES

BACKGROUND

The Emergency Feedwater (EFW) system provides an independent safety related means of supplying feedwater to the steam generators for removal of decay heat and prevention of reactor core uncovery during emergency phases of plant operation. The EFW system is a dedicated safety system which has no operating functions for normal plant operation. It does this by supplying water from the Emergency Feedwater Storage Tank (EFWST), covered by LCO 3.7.5, to the steam generator secondary side via a connection to the main feedwater piping inside containment. Steam is released to the atmosphere from the steam generators via the main steam safety valves or atmospheric dump valves.

Signal

Automatic EFW System actuation on low steam generator level is accomplished by the Emergency Feedwater Actuation ~~System~~ (EFAS) or the Alternate Protection System (APS). The EFAS will actuate feedwater to either or both steam generators with low levels, and will terminate EFW to a steam generator having a significantly high steam generator level.

Alternate Feedwater Actuation Signal (AFAS) of the

of the Engineered Safety Features Actuation System (ESFAS)

The EFW System is configured into two separate mechanical trains. Each train is aligned to feed its respective steam generator. Each train consists of one Emergency Feedwater Storage Tank (EFWST), one 100% capacity motor-driven pump subtrain, one 100% capacity steam-driven pump subtrain, valves, one cavitating venturi, and specified instrumentation. Each pump subtrain takes suction from its respective EFWST and has its respective discharge header. Each subtrain discharge header contains a pumps discharge check valve, flow regulating valve, steam generator isolation valve and steam generator isolation check valve. The motor-driven subtrain and steam-driven subtrain are joined together inside containment to feed their respective steam generator through a common EFW header which connects to the steam generator downcomer feedwater line. Each common EFW header contains a cavitating venturi to restrict the maximum EFW flow rate to each steam generator. The cavitating venturi restricts the magnitude of the two pump flow as well as the magnitude of individual pump runout flow to the steam generator.

(continued)

BASES

BACKGROUND (continued)

~~A cross-connection is provided between each EFWST so that either tank can supply either train of EFW.~~ Pump discharge crossover piping is provided to enhance system versatility during long-term emergency modes, such that a single pump can feed both steam generators. Two normally locked closed, local manually operated isolation valves are provided for subtrain separation.

One-hundred percent capacity is sufficient to remove decay heat and cool the plant to shutdown cooling entry conditions at the design cooldown rate, 75°F/hr. Fifty percent capacity is sufficient to remove decay heat but is insufficient to maintain the design cooldown rate. The diverse motive power of the two trains meets the diversity requirement of BTP ASB 10-1 (Ref. 4).

The EFW System is one of the systems required to meet GDC 34 (Ref. 2.a) and GDC 44 (Ref. 2.b) regarding the capability to remove decay heat and transfer it to an ultimate heat sink, in this case the atmosphere.

An OPERABLE EFW System is required if the steam generators are to be considered OPERABLE.

The EFW System is discussed in Section 10.4.9 of the CESSAR-DC (Ref. 1).

APPLICABLE SAFETY ANALYSES

The EFW System is designed to supply enough water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the Main Steam Safety Valves (MSSVs). Subsequently, the EFW System supplies sufficient water to cool the plant to shutdown cooling entry conditions with steam being released through the ADVs.

The EFW System design must be such that it can perform its function following a feedwater line break between the MFIV and containment, disabling EFW supply to one steam generator, with a loss of offsite power following turbine trip, and a single active failure of the steam-turbine driven EFW Pump. In such a case, one steam generator is lost for heat removal but the other steam generator still can provide the required heat removal capability. The EFW System design meets the single failure criterion of BTP ASB 10-1 (Ref. 4).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Note that a pipe break in the EFW System can lead to a Loss of Feedwater event in addition to disabling EFW to one steam generator. The EFW System must still meet single failure criteria, see BTP ICSB-13 (Ref. 8). See also SRP 3.6.1 (Ref. 9) for pipe breaks outside containment.

The EFW System meets Criterion 3 for inclusion as a technical specification because it is on the primary success path for RCS heat removal for all events in which the steam generators transfer heat from the RCS. Even for a large LOCA reflux cooling (by condensing steam) in the steam generator tubes is possible.

LCOs

The LCO ensures the availability of at least one steam generator to remove residual heat for all events accompanied by a loss of offsite power and single failure. This is accomplished by two redundant and diverse emergency feedwater pumps for each steam generator.

APPLICABILITY

Because the EFW System is the safety grade means of removing core heat, it must be OPERABLE whenever the steam generators are required for RCS heat removal in MODES 1 through 4.

In MODES 5 and 6, the steam generators are not normally used for decay heat removal and the EFW System is not required.

ACTIONS

A.1

With one of the required EFW pumps inoperable, action must be taken to restore the pump to OPERABLE status. The 72 hour Completion Time is similar to that for ECCS systems for which it has been shown to be a suitable limit on risk.

B.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. Six hours is a reasonable time, based on operating experience, to reach MODE 3 from full power conditions without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

This action also commences with two EFW pumps inoperable, to bring the plant to a mode in which alternative means of decay heat removal are available via the Main or Startup Feedwater system.

Thirty Six hours is a reasonable time, based on operating experience, to reach MODE 5 from full power conditions without challenging plant systems. This places the plant in a MODE in which the LCO is not applicable.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

Verifying the correct alignment for manual, power operated, and automatic valves in the EFW flowpath provides assurance that the proper flowpaths exist for EFW operation. This SR does not apply to valves which are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves which cannot be inadvertently misaligned, such as check valves. The 31-day frequency is based on engineering judgement considering the importance of these valves and the low probability of their misalignment.

SR 3.7.4.2

This SR demonstrates that the EFW pumps develop sufficient discharge pressure to deliver the required flow at the full open pressure of the MSSVs. Because it is undesirable to introduce cold EFW into the Steam Generators while they are operating, this testing is performed on recirculation flow. Periodically comparing the reference differential pressure developed at this reduced flow detects trends that might be indicative of incipient failures. The ASME Section XI (Ref. 3) inservice testing (only required at three month intervals) satisfies this requirement when performed, per Specification 6.8.1.j, Inservice Inspection and Testing program. A 31-day frequency on a STAGGERED TEST BASIS results in testing each pump once per three months as required by the ASME code.

SR 3.7.4.2 is modified by a Note to allows an exception to SR 3.0.4. Provisions of SR 3.0.4 are not applicable for entry into MODE 3 for purposes of testing the turbine driven EFW pumps due to insufficient amount of steam in MODES 4, 5, and 6 to perform a valid test.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.4.3

This SR ensures that EFW can be delivered to the appropriate Steam Generator in the event of any accident or transient that generates an EFAS by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. Although the actuation logic is tested as part of the SIAS functional test every 92 days, the subgroup relays that actuate the system cannot be tested during normal plant operation. The surveillance interval of 18 months is based on the refueling cycle, and has been shown to be acceptable through operating experience.

SR 3.7.4.4

This SR ensures that the EFW pumps will start in the event of any accident or transient that generates an EFAS by demonstrating that each EFW pump starts automatically on an actual or simulated actuation signal. Although the actuation logic is tested as part of the SIAS functional test every 92 days, the subgroup relays that actuate the system cannot be tested during normal plant operation. The surveillance interval of 18 months is based on the refueling cycle, and has been shown to be acceptable through operating experience.

SR 3.7.4.4 is modified by a Note to suspend the provisions of SR 3.0.4 for entry into MODE 3 for purposes of testing the turbine driven EFW pumps due to insufficient amount of steam in MODES 4, 5, and 6 to perform a valid test.

SR 3.7.4.5

This SR ensures that the EFW system is properly aligned by demonstrating the flowpath to each Steam Generator prior to entering MODE 2 operation, after > 30 days in MODE 5 or 6. Operability of EFW flow paths must be demonstrated before sufficient core heat is generated requiring operation of the EFW system during a subsequent shutdown. The frequency is based on the probability of improper valve lineups occurring during an extended outage. This SR ensures that the flow path from the EFWST to the Steam Generators is properly aligned by requiring a verification of flow capacity of at least 500 gpm at 1270 psia. In addition, this SR allows for flexibility by permitting testing at other flow rates and pressures that demonstrate equivalent flow.

(continued)

BASES

REFERENCES

1. CESSAR-DC Section 10.4.9, Emergency Feedwater System.

2. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Standard Technical Specifications - Volume 1 (Criteria Application)."

3. American Society for Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400 "Inservice Tests - Category A and B Valves."

Additional References

4. 10 CFR 50, Appendix A, GDC 34 - Residual Heat Removal.

5. 10 CFR 50, Appendix A, GDC 44 - Cooling Water.

6. Branch Technical Position RSB 5-1, "Design Requirements for the Residual Heat Removal System," attached to SRP Section 5.4.7.

7. Branch Technical Position ASB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants," attached to SRP Section 1 5.9.

8. NRC Generic Letter 88-03, "Resolution of Generic Safety Issue 93, 'Steam Binding of Auxiliary Feedwater Pumps'" February 17, 1988.

9. NUREG-0800, "Standard Review Plan", (SRP) Section 10.4.9, Rev. 2, "Auxiliary Feedwater System (PWR)."

10. BTP ICSB 13, "Design Criteria for Auxiliary Feedwater Systems," Appendix A to SRP Section 10.3 "Main Steam Supply System," Rev. 3, April 1984.

11. NUREG-0800, "Standard Review Plan", Section 3.6.1 "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

(continued)

BASES

REFERENCES (continued)

Additional References (continued)

12. RG 1.139, "Guidance for Residual Heat Removal", May 1978.

13. ANS-51.10-1979 "Auxiliary Feedwater System for Pressurizer Water Reactors."

14. ANS-58.11-1983 "Cooldown Criteria for Light Water Reactors."

15. NUREG 0737 "Clarification of TMI Action Plan," October 31, 1980.

16A.10.5 B 3.7.5 EMERGENCY FEEDWATER STORAGE TANK

B 3.7 PLANT SYSTEMS

B 3.7.5 Emergency Feedwater Storage Tank

BASES

BACKGROUND

The Emergency Feedwater Storage Tank (EFWST) provides a safety grade source of water for removing decay and sensible heat from the Reactor Coolant System (RCS) during emergency phases of the plant. The EFWST provides a passive flow of water by gravity to the emergency feedwater (EFW) pumps. The EFW pumps supply this water to the steam generators to remove heat from the RCS. The steam produced is released to the atmosphere by the main steam safety valves (MSSVs) or the atmospheric dump valves (ADVs).

When the main steam isolation valves (MSIVs) are open, the preferred means of heat removal is to discharge steam to the condenser by the non-safety grade path of the steam bypass valves. This has the advantage of conserving condensate while minimizing releases to the environs.

There are two EFWSTs, each tank with one motor driven and one steam driven feedpump is assigned for each steam generator.

A normal locked closed, local manually operated isolation valve is provided for each EFWST to provide separation. A line connected to a non-safety source of condensate is also provided with local manual isolation so that it can be manually aligned for gravity feed to either of the EFWSTs, should the EFWSTs reach low level before Shutdown Cooling System entry conditions are reached.

Each tank contains 100% of the total required water supply.

A description of the EFWST is found in CESSAR-DC Section 10.4.9 (Ref. 1). Because the EFWST is a principal component in removing residual heat from the RCS, it is designed to withstand earthquakes and other natural phenomena, as well as missiles which might be generated by natural phenomena.

The water volume of each EFWST [350,000 gallons] is determined by the quantity required to achieve safe cold shutdown considering:

(continued)

BASES

BACKGROUND (continued)

1. A main feedline break without isolation of EFW flow to the affected steam generator for 30 minutes.
2. Refill of the intact steam generator
3. Eight hours of operation at hot standby conditions
4. Subsequent cooldown of RCS within six hours to conditions which permit operation of the shutdown cooling system
5. Continuous operation of one reactor coolant pump.

At the end of this cooldown, the EFWST level must be sufficient to ensure adequate NPSH for the operating EFW pumps.

The EFWST is one of the systems required to meet GDC 34 (Ref. 2.a) and GDC 44 (Ref. 2.b) regarding the capability to remove decay heat and transfer it to an ultimate heat sink.

APPLICABLE SAFETY ANALYSES

The EFWST provides cooling water to remove decay heat and cooldown the plant following all events in the accident analysis, CESSAR-DC Chapters 6 and 15. For anticipated operating occurrences and accidents which do not affect the operability of the steam generators, the analysis assumption is generally 30 minutes at MODE 3, steaming through the MSSVs, followed by a cooldown to shutdown cooling entry conditions at the design cooldown rate.

The CESSAR-DC Chapters 6 and 15 accident analysis does not form the basis for the EFWST volume as the events analyzed require less condensate than the design basis. The limiting event for the condensate volume is the large feedwater line break with a loss of offsite power. Single failures that also affect this event include 1) the failure of the diesel generator powering the motor driven EFW Pump to the unaffected steam generator (requiring additional steam to drive the remaining EFW Pump's turbine), and 2) the failure of the steam driven EFW Pump. These are not usually the limiting failures in terms of consequences for these events.

The EFWST meets Criterion 3 for inclusion as a technical specification because it is in the primary success path for RCS heat removal for all events in which the steam generators are available for heat removal from the RCS.

(continued)

BASES

LCOs

To satisfy accident analysis assumptions, the EFWST must contain sufficient cooling water to remove decay heat for 30 minutes following a reactor trip from 102% RATED THERMAL POWER and then cooldown the RCS to shutdown cooling entry conditions, assuming a loss of offsite power and the most adverse single failure. In doing this it must retain sufficient water to ensure adequate NPSH for the EFW pumps during the cooldown, as well as to account for any losses from the steam driven EFW Pump's turbine, or before isolating EFW to a broken line.

The level specified ([LATER]) equates to a usable volume of [350,000] gallons, which is based on holding the plant in MODE 3 for eight hours followed by a cooldown to Shutdown Cooling entry conditions at 75°F/hour. This bases is established by BTP RSB 5-1 (Ref. 3) and exceeds the volume required by the accident analysis.

APPLICABILITY

The required condensate volume must be available whenever the steam generators provide the heat sink for the RCS. Once a cooldown commences, the condensate volume may be reduced by using it for the cooldown. Proceeding with the cooldown ensures that the plant can reach shutdown cooling entry conditions on the available condensate inventory. A lesser condensate volume in the EFWST is required in MODES 2, 3 and 4 than in MODE 1 since the mass of fluid in the steam generators is greater at zero power than at full power.

~~No requirements are placed on the EFWST during MODE 4 as this is the transition MODE to Shutdown Cooling System operation.~~

In MODES 5 and 6 the steam generators are not required for cooldown, and the inventory in the EFWST is not required.

(continued)

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BASES

ACTIONS

A.1

With the EFWST(s) unable to supply the required volume of cooling water to the EFW pumps, it must be restored to OPERABLE status. Four hours allows time to restore the required volume from the backup supply. Four hours is a reasonable time to limit the risk from accidents and AOOs requiring the plant to cool down. With the level slightly below that required, there is still sufficient inventory to conduct a cooldown, although a cooldown should start immediately, should the condenser become unavailable as a heat sink.

A.2.1 and A.2.2

As an alternative to shutting down the unit, verification that the other EFWST is operable may be done before 4 hours expires. In such a case, the EFWST must still be returned to OPERABLE status within 72 hours.

B.1 and B2

When a Required Action cannot be completed within the Completion Time, a controlled shutdown should be commenced. Six hours is a reasonable time, based on operating experience, to reach MODE 3 from full power conditions without challenging plant systems.

Continuing the plant shutdown begun in Required Action B.1, 12 hours is a reasonable time, based on operating experience, to reach MODE 4 from full power conditions without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Checking the EFWST level verifies that the EFWST contains the required volume of cooling water. Checking once per 12 hours is adequate because the operator should be aware of plant evolutions which can affect the EFWST inventory between checks.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.2

Checking the other EFWST supply volume verifies it contains the required volume of cooling water. Checking once per 12 hours is adequate because the operator should be aware of plant evolutions which can affect the inventory between checks.

REFERENCES

1. CESSAR-DC Section 10.4.9, Emergency Feedwater System.

2. Title 10 Code of Federal Regulations Part 50 (10 CFR 50), Appendix A, General Design Criteria:

   a. GDC 34 - Residual Heat Removal.
   b. GDC 44 - Cooling Water.

3. Regulatory Guide 1.139, "Guidance for Residual Heat Removal", May 1978.

4. ANS-58.11-1983 "Cooldown Criteria for Light Water Reactors".

5. Branch Technical Position RSB 5-1, "Design Requirements for the Residual Heat Removal System".

16A.10.6 B 3.7.6 SECONDARY SPECIFIC ACTIVITY

B 3.7 PLANT SYSTEMS

B 3.7.6 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from Steam Generator tube out-leakage from the Reactor Coolant System (RCS). Under steady-state conditions, the activity is primarily Iodines with relatively short half lives, and thus is indicative of current conditions. During transients, Xenon isotopes may be significant. Isotopes, other than these halogens and noble gases, from activated corrosion products may appear.

A limit on Secondary Activity during power operation minimizes releases to the environs because of normal operation, anticipated operational occurrences, and accidents.

3.4.12

The LCO limit is lower than the activity value which might be expected from a 0.5 gpm tube leak, LCO ~~3.3.11~~, of primary coolant at the limit of 1.0 μCi/gram, LCO ~~3.3.12~~ 3.4.15. For example, such a leak could increase Secondary Activity by about 0.015 μCi/gram (for a steam generator at zero power with a mass of approximately 292,000 lbm) per 24 hours of leakage. This is offset by secondary blowdown, RCS purification, and radioactive decay of the isotopes. Most of the Iodine isotopes have short half lives, i.e., less than 20 hours. I-131 with a half life of 8.04 days concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.

With the specified activity limit, the resultant two-hour thyroid dose to a person at the exclusion area boundary would be about $\boxed{.13\ \text{Rem}}$ should the MSSVs open for the two hours following a trip from full power. This example is based on:

- 28,600 lbm released at an average concentration of 0.1 μCi/gram,
- Decontamination Factor (DF) of 10,
- Dose Conversion factor of 1.48 E6 Rem/Ci,
- Atmospheric dispersion factor: χ/Q = 1.00 E-3 sec/cubic meter,
- Breathing rate of 3.47 E-4 cubic meter/sec for a person at the exclusion area boundary.

(continued)

BASES

BACKGROUND (continued)

From these examples, it can be seen that operating the plant at the allowable limits results in a 2-hour exclusion area boundary (EAB) exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES

The accident analysis of the MSLB failure (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.1 μCi/g DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of a MSLB do not exceed a small fraction of the plant Exclusion Area boundary limits of 10 CFR 100 for whole body and thyroid dose rates.

With the loss of offsite power, the remaining Steam Generator is available for core decay heat dissipation by venting steam to the atmosphere through the main steam safety valves (MSSVs) and Steam Generator atmospheric dump valves (ADVs). The Auxiliary Feedwater System supplies the necessary makeup to the Steam Generator. Venting continues until the reactor coolant temperature and pressure has decreased sufficiently for the Shutdown Cooling System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the Steam Generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected Steam Generator is assumed to discharge steam and any entrained activity through the MSSVs and ADVs during the event. Since no credit is taken in the analysis for activity plate-out or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary Activity satisfies the requirements of Criterion 3 of the Interim Policy Statement as described in Reference 3.

(continued)

BASES

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of 0.1 $\mu Ci/g$ DOSE EQUIVALENT I-131 is required to contain the radiological consequences of a DBA to a small fraction of 10 CFR 100.

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3 and 4 the limits on secondary activity apply whenever using the Steam Generators for RCS heat removal. This is the time of potential secondary steam releases to atmosphere, carrying with the steam a portion of the activity in the Steam Generators.

In MODES 5 and 6, the Steam Generators are not being used for heat removal. Both the RCS and Steam Generators are depressurized, and primary to secondary leakage is minimal. Therefore, monitoring of secondary activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value is an indication of a problem in the RCS, as well as contributing to increased post-accident doses. The plant should be shut down in an orderly manner to minimize the increased DOSE EQUIVALENT I-131 in the RCS, potentially increasing the secondary activity even further. An orderly shutdown also minimizes potential releases to the environs. The plant must be placed in a MODE in which the requirement does not apply if Secondary Activity cannot be restored to within limits in the associated Completion Time. This is done by placing the plant in at least MODE 3 in six hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power operation without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR ensures that the Secondary Activity is within the limits of the accident analysis. A Gamma Isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the accident analysis assumptions as to the source terms in post-accident releases. It also serves to identify and trend any unusual isotopic concentrations which might indicate changes in reactor coolant activity or leakage. The 31 day frequency allows the level of DOSE EQUIVALENT I-131 to be monitored, increasing trends to be detected, and appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 100, "Site Dose Criteria".

2. CESSAR-DC Section 15.1, Main Steam Line Break Accident Analysis Radioactivity Release Methodology.

3. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Standard Technical Specifications - Volume 1 (Criteria Application)."

Additional References

4. NUREG-0800, Standard Review Plan, Section 15.1.5, Steam System Piping Failures inside and Outside of Containment.

5. 10 CFR 50, Appendix A, GDC 60 - Control of Releases of Radioactive Materials to the Environment.

6. CESSAR-DC Chapter 15, Radioactivity Release Methodology.

16A.10.7 B 3.7.7 COMPONENT COOLING WATER

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water

BASES

BACKGROUND

The Component Cooling Water System (CCWS) is a closed loop cooling water system which cools components and heat exchangers ~~located in the Auxiliary, Fuel, Radwaste, and Containment Buildings~~. The CCWS is capable of removing sufficient heat using various combinations of pumps and heat exchangers to:

connected to the CCWS.

(1) Ensure a safe reactor shutdown coincident with loss of offsite power
(2) Perform a normal shutdown cooling of the reactor within 24 hours

(3) Perform a safety grade shutdown cooling of the reactor within 36 hours

(4) Perform Post LOCA cooling

(5) Perform normal power operation cooling.

The CCWS consists of two separate, independent, redundant, closed loop, safety related divisions. Either division of the CCWS is capable of supporting 100% of the cooling functions required for a safe reactor shutdown.

Each division of the CCWS includes two heat exchangers, a surge tank, two component cooling water pumps, a chemical addition tank, a component cooling water radiation monitor, two sump pumps, a component cooling water heat exchanger ~~Building~~ structure sump pump, piping, valves, controls, and instrumentation. No cross connections between the two divisions exist. ~~Each division consists of an essential and non essential cooling loop.~~

The CCWS provides cooling water to essential and non-essential components.

~~Cooling to the spent fuel pool heat exchanger(s) and the non-essential loop is isolated on a SIAS.~~ If these headers fail to isolate, the idle component cooling water pump in the respective loop will automatically start on a low pump differential pressure signal. This assures that there is no flow degradation to the safety related components. The ~~non safety related heat loads~~ non-essential headers and the RCP ~~heat loads~~ headers isolate on a ~~low~~ low-low surge tank level.

The non-essential headers and the spent fuel pool cooling heat exchangers are isolated automatically on an SIAS.

(continued)

BASES

BACKGROUND (continued)

Makeup water to the CCWS is normally supplied by the Demineralized Water System. The backup makeup water source is from the Station Service Water System (SSWS).

The CCWS serves as an intermediate cooling water system between the Reactor Coolant System (RCS) and the SSWS. A radiation monitor is provided at the outlet of the component cooling water ~~heat exchangers~~ Pumps to detect any radioactive leakage into the CCWS that may contain radioactivity.

Additional information on the design and operation of the system, along with a list of components served, can be found in CESSAR-DC Section 9.2.2 (Reference 1).

APPLICABLE SAFETY ANALYSES

The CCWS, in conjunction (UHS) with the Station Service Water System (SSWS) and the ultimate heat sink, is capable of removing sufficient heat from the essential heat exchangers to ensure a safe reactor shutdown and cooling following a postulated accident coincident with a loss of offsite power.

The CCWS, in conjunction with the SSWS, is capable of maintaining the outlet temperature of the ~~CCWS~~ component cooling water heat exchanger within the limits of 65° and 120°F during a design basis accident with loss of offsite power.

A single failure of any component in the CCWS will not impair the ability of the CCWS to meet its functional requirements.

Delete and replace with

The CCWS also functions to cool the plant from 210°F shutdown cooling entry conditions (T_{cold} < 350°F) to MODE 5 (T_{cold} < ~~200°F~~ 210°F) during ~~normal and~~ post-accident operations. The time required ~~to cool from~~ 350°F to ~~200~~ 210°F is a function of the number of CCWS and Shutdown Cooling System divisions operating. One division with one pump and heat exchanger is sufficient to remove decay heat during subsequent operations with T_{cold} < ~~200~~ 210°F.

The CCWS satisfies the requirements of Criterion 3 of the Interim Policy Statement as described in Reference 2.

(continued)

The CCWS, in conjunction with the SCS and SSWS, is designed to cool the reactor coolant from 350°F to 140°F through the shutdown cooling heat exchangers and the component cooling water heat exchangers. The reactor can be cooled to 140°F within 24 hours after reactor shutdown by first cooling the reactor coolant to 350°F through the steam generators and then cooling to 140°F by utilizing both divisions of SCS, CCWS, and SSWS.

BASES

LCO

The CCWS divisions are completely 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 division of CCWS is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two divisions of CCWS must be OPERABLE. At least one division will operate assuming the worst single active failure occurs coincident with the loss of off-site power.

A division is considered OPERABLE when:

1. it has an OPERABLE pump and associated surge tank, and

2. the associated piping, valves, heat exchanger and instrumentation on the essential flowpath are OPERABLE.

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

APPLICABILITY

In MODES 1, 2, 3, and 4 the CCW system is a normally-operating system, which must be available to perform its post-accident safety functions, primarily RCS heat removal by cooling the Shutdown Cooling Heat Exchanger.

In MODES 5 and 6, the OPERABILITY requirements of the CCW System are determined by the systems its supports.

ACTIONS

A.1 and A.2

With one CCWS division inoperable, a review of the systems and components provided in Reference 1 must ensure that all Safety Related systems or components which are supported by the OPERABLE CCWS division are also OPERABLE. This is a cross division check to confirm that redundant systems or components are not rendered inoperable due to the loss of a single CCW division.

(continued)

BASES

ACTIONS (continued)

The Completion Time of four hours is intended to allow the operator time to evaluate and restore any discovered inoperabilities. Also, in this Required Action, the Completion Time only begins upon discovery that both 1) an inoperable CCWS division exists, and 2) a required system or component on the other CCWS division is inoperable.

The Completion Time is based on engineering judgment taking into consideration the probability of an event requiring the function of an inoperable system occurring while in this condition.

With one CCWS division inoperable, the inoperable CCWS division must be restored to OPERABLE status within 72 hours. In this condition, the remaining OPERABLE CCWS division is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE division could result in reduced heat removal capability. The 72 hour Completion Time is based on the heat removal function and is consistent with the engineered safety features systems' Completion Time for loss of one redundant train.

B.1 and B.2

The plant must be placed in a MODE in which the requirement does not apply if the CCWS division cannot be restored to OPERABLE status in the associated Completion Time. This is done by placing the plant in at least MODE 3 in 6 hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power operation without challenging plant systems.

C.1, C.2 and C3

With both divisions inoperable, action must be taken to restore at least one division to OPERABLE status. In this case there is no heat sink for the Shutdown Cooling System and the plant should be placed in a condition where decay heat can be removed by the Steam Generators. The time allowed (12 hours) is reasonable based on operating experience, to place the plant in MODE 4 from full power conditions without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

With both divisions inoperable, flexibility is left to the operator (and abnormal operating procedures) to manage the situation. This allows remaining in MODE 4 with an alternate means of heat removal. If an adequate complement of CCW components is available the plant should be placed in MODE 5.

This Action allows total loss of function without entry into LCO 3.0.3 because entry into MODE 5, as required by LCO 3.0.3 may not be possible with two CCW division inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flowpath provides assurance that the proper flowpaths exist for CCW operation. This SR does not apply to valves which are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves which cannot be inadvertently misaligned, such as check valves. The 31-day /frequency is based on engineering judgement considering the importance of these valves and the low probability of their misalignment.

SR 3.7.7.2

This SR demonstrates proper automatic operation of the CCW valves. The CCWS is a normally-operating system. The 18-month frequency was developed considering it was prudent that many surveillances only be performed during a plant outage. This was due to the plant conditions needed to perform the SR and the potential for unnecessary plant transients if the SR is performed with the reactor at power. Operating experience has shown these components virtually always pass the SR when performed in the 18-month frequency. Therefore, the frequency was concluded to be acceptable from a reliability standpoint. NRC Generic Letter 83-27 (Ref. 3) affirmed these conclusions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.7.3

This SR demonstrates proper automatic operation of the CCW pumps. The CCW System is a normally-operating system. The 18-month frequency was developed considering it was prudent that many surveillances only be performed during a plant outage. This was due to the plant conditions needed to perform the SR and the potential for unnecessary plant transients if the SR is performed with the reactor at power. Operating experience has shown these components virtually always pass the SR when performed on the 18-month frequency. Therefore, the frequency was concluded to be acceptable from a reliability standpoint. NRC Generic Letter 83-27 (Ref. 3) affirmed these conclusions.

REFERENCES

1. CESSAR-DC Section 9.2.2, Component Cooling Water System.

2. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Standard Technical Specifications - Volume 1 (Criteria Application)."

3. NRC Generic Letter No. 83-27, "Surveillance Intervals in Standard Technical Specifications."

4. CESSAR-DC Section 6.2, Containment Analysis.

5. CESSAR-DC Section 5.4, Residual Heat Removal.

6. 10 CFR 50, Appendix A, GDC 34 - Residual Heat Removal.

7. 10 CFR 50, Appendix A, GDC 38 - Containment Heat Removal.

8. 10 CFR 50, Appendix A, GDC 44 - Cooling Water.

9. 10 CFR 50, Appendix A, GDC 45 - Inspection of Cooling Water Systems.

10. 10 CFR 50, Appendix A, GDC 46 - Testing of Cooling Water Systems.

(continued)

BASES

REFERENCES (continued)

11. ANSI/ANS 59.1-1979, "Safety Related Cooling Water Systems in Nuclear Power Plants".

16A.10.8 B 3.7.8 STATION SERVICE WATER SYSTEM

B 3.7 PLANT SYSTEMS

B 3.7.8 Station Service Water System

BASES

BACKGROUND

The Station Service Water System (SSWS) provides a heat sink for the removal of process and operating heat from safety related components during a transient or DBA through the CCWS. During normal operation, and a normal shutdown, the ~~SWS~~ SSWS also provides this function for various safety related and non-safety related components through the CCWS.

The SSWS consists of two separate, redundant, open loop, safety related divisions. Each division cools one of two divisions of the CCWS, which in turn cools 100% of the safety-related loads. The SSWS operates at a lower pressure than the CCWS to prevent contamination of the CCWS with raw water.

Each division of the SSWS consists of two pumps, two strainers, two sump pumps, and associated piping, valves, controls and instrumentation. ~~Each SSWS division pumps are located in a separate SSWS pump structure.~~ The station service water pumps circulate cooling water to the component cooling water heat exchanger and back to the ultimate heat sink. Provisions are made to ensure a continuous flow of cooling water under normal and accident conditions.

Additional information about the design and operation of the ~~SWS,~~ SSWS along with a list of the components served, can be found in CESSAR-DC Section 9.2.1 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The SSWS, in conjunction with the Component Cooling Water System (CCWS) and ultimate heat sink (UHS), is capable of removing sufficient heat to ensure a safe reactor shutdown coincident with a loss of offsite power.

The SSWS is capable of maintaining the CCWS supply temperature of 120°F or less following the design basis accident under the most adverse historical meteorological conditions consistent with the intent of Regulatory Guide 1.27.

(continued)

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December 21, 1990

BASES

APPLICABLE SAFETY ANALYSES (continued)

A single failure of any component in the SSWS will not impair the ability of the SSWS to meet its functional requirements.

The SSWS, in conjunction with the CCWS and SCS, is designed to cool the reactor coolant from 350°F to 140°F through the shutdown cooling heat exchangers and the component cooling water heat exchangers. The reactor coolant system can be cooled to 140°F within 24 hours after reactor shutdown by utilizing both divisions of the SCS, CCWS, and SSWS.

first cooling the reactor coolant to 350°F through the steam generators and then cooling the reactor coolant to 140°F by

The SSWS, in conjunction with the CCWS, is designed to provide a maximum component cooling water temperature of 105°F or less during normal operating modes.

normal chilled water condensers

The SSWS through the CCWS is designed to provide cooling water to the RCPs, letdown heat exchanger, ~~nuclear~~ sample ~~coolers, non-essential chillers,~~ and other ~~non-safety~~ non-essential reactor auxiliary cooling loads. heat exchangers

The SSWS satisfies the requirements of Criterion 3 of the Interim Policy Statement as described in Reference 2.

LCO

Two SSWS divisions provide the required redundancy to ensure the system functions to remove post-accident heat loads, assuming the worst single active failure occurs coincident with the loss of off-site power.

A division is considered OPERABLE when:

1. it has an OPERABLE pump, and

2. the associated piping, pump valves, instrumentation, heat exchanger strainer, and SSW~~S~~ structure ventilation fans on the safety related flowpath are OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4 the SSWS system is a normally-operating system, which must be available to perform its post-accident safety functions, primarily RCS heat removal by cooling the CCWS Heat Exchanger.

(continued)

BASES

APPLICABILITY (continued)

In MODES 5 and 6, the OPERABILITY requirements of the SSWS are determined by the systems it supports. Since no specifications for SSWS is provided for MODES 5 and 6, the definition of OPERABILITY applies to the systems supported by SSWS (i.e., LCO 3.0.7 does not apply to SSWS in MODES 5 and 6).

ACTIONS

A.1

With one SSWS division inoperable, a review must ensure that the CCWS division supported by the OPERABLE SSWS division is also OPERABLE. This is a cross division check to confirm that redundant systems or components are not rendered inoperable due to the loss of a single SSWS division.

The Completion Time of four hours is intended to allow the operator time to evaluate and restore any discovered inoperabilities. Also, in this Required Action, the Completion Time only begins upon discovery that both 1) an inoperable SSWS division exists, and 2) the CCWS division supported by the other SSWS division is inoperable.

The Completion Time is based on engineering judgment taking into consideration the probability of an event requiring the function of an inoperable system occurring while in this condition.

In addition, the inoperable SSWS division must be restored to OPERABLE status within 72 hours. In this condition, the remaining OPERABLE SSWS division is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE division could result in reduced heat removal capability. The 72 hour Completion Time is based on the heat function and is consistent with the engineered safety feature systems' Completion Time for loss of one redundant division.

(continued)

ACTIONS (continued)

B.1 and B.2

The plant must be placed in a MODE in which the requirement does not apply if the SSWS division cannot be restored to OPERABLE status in the associated Completion Time. This is done by placing the plant in at least MODE 3 in 6 hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power operation without challenging plant systems.

C.1 and C.2

With both divisions inoperable, action must be taken to restore at least one division to OPERABLE status. In this case there is no heat sink for the Shutdown Cooling System and the plant should be placed in a condition where decay heat can be removed by the Steam Generators. The time allowed (12 hours) is reasonable based on operating experience, to place the plant in MODE 4 from full power conditions without challenging plant systems.

With both divisions inoperable, flexibility is left to the operator (and abnormal operating procedures) to manage the situation. This allows remaining in MODE 4 with an alternate means of heat removal. When an adequate complement of SWS components is available the plant should be placed in MODE 5.

This allows total loss of function without entry into LCO 3.0.3 because entry into MODE 5, as required by LCO 3.0.3, may not be possible with two SSWS divisions inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Verifying the correct alignment for manual and motor operated, valves in the SSWS flowpath provides assurance that the proper flowpaths exist for SSWS operation. This SR does not apply to valves which are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves which cannot be inadvertently misaligned, such as check valves.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

The 31-day frequency is based on engineering judgement considering the importance of these valves and the low probability of their misalignment.

SR 3.7.8.2

This SR demonstrates proper automatic operation of the SSWS pumps. The SSWS is a normally-operating system. The 18-month frequency was developed considering it was prudent that many surveillances only be performed during a plant outage. This was due to the plant conditions needed to perform the SR and the potential for unnecessary plant transients if the SR is performed with the reactor at power. Operating experience has shown these components virtually always pass the SR when performed on the 18-month frequency. Therefore, the frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. CESSAR-DC Section 9.2.1, Station Service Water System.

2. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Standard Technical Specifications - Volume 1 (Criteria Application)."

Additional References

3. CESSAR-DC Section 6.2, Containment Analysis.

4. CESSAR-DC Section 5.4.7, Residual Heat Removal.

5. 10 CFR 50, Appendix A, GDC 34 - Residual Heat Removal.

6. 10 CFR 50, Appendix A, GDC 38 - Containment Heat Removal.

7. 10 CFR 50, Appendix A, GDC 44 - Cooling Water.

8. 10 CFR 50, Appendix A, GDC 45 - Inspection of Cooling Water Systems.

(continued)

BASES

REFERENCES (continued)

9. 10 CFR 50, Appendix A, GDC 46 - Testing of Cooling Water Systems.

10. ANSI/ANS 59.1-1979, "Safety Related Cooling Water Systems in Nuclear Power Plants."

16A.10.9 B 3.7.9 ULTIMATE HEAT SINK

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink

BASES

BACKGROUND

The Ultimate Heat Sink (UHS) provides a heat sink for process and operating heat from safety-related components during a transient or accident, as well as during normal operation. This is done utilizing the Station Service Water System (SSWS) and the Component Cooling Water (CCWS) System.

The Ultimate Heat Sink consists of the [cooling water pond] associated piping, valves, and instrumentation. Additional information on the design and operation of the system along with a list of components served can be found in CESSAR-DC Section 9.2.5 (Ref. 1).

If the UHS does not meet its design limits (water temperature, water level), the UHS may not have sufficient capacity to bring the plant to a safe controlled shutdown during a DBA from full power, but may be able to support plant operation at a reduced power level.

APPLICABLE SAFETY ANALYSES

Condenser

The UHS removes heat from the reactor core following all accidents and Anticipated Operational Occurrences (AOOs) in which the plant is cooled down and placed on shutdown cooling. For those plants using it as the normal heat sink for condenser cooling via the Circulating Water System, plant operation ~~at full power is its maximum~~ heat load. Its maximum post-accident heat load occurs following a design basis Loss Of Coolant Accident (LOCA), when recirculation of the IRWST through the Containment ~~Cooling~~ Spray System~~s~~ is required to remove the core decay heat.

The operating limits are based on a conservative heat transfer analyses for the worse case LOCA. Refer to CESSAR-DC Section 9.2.5 (Ref. 1) for details of the assumptions used in the analysis. These assumptions include: worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst case single active failure. The UHS is designed in accordance with Regulatory Guide 1.27 (Ref. 2) which requires a 30-day supply of cooling water in the UHS.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The Ultimate Heat Sink satisfies the requirements of Criterion 3 of the Interim Policy Statement as described in Reference 3.
LCO	The UHS is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature which would allow the SSWS to operate for at least 30 days following the design basis LOCA without the loss of NPSH and without exceeding the maximum design temperature of the equipment served by the SSWS. To meet this condition the UHS temperature should not exceed 95°F and the level should not fall below [site specific] during normal plant operation.
APPLICABILITY	In MODES 1, 2, 3 and 4 the UHS is normally-operating and must be prepared to perform its post-accident safety functions, primarily RCS heat removal. In MODES 5 and 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.
ACTIONS	A.1 and A.2 The plant must be placed in a MODE in which the requirement does not apply if the UHS cannot be restored to OPERABLE status in the associated Completion Time. This is done by placing the plant in at least MODE 3 in 6 hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power operation without challenging plant systems.
SURVEILLANCE REQUIREMENTS	SR 3.7.9.1 This SR ensures adequate long term (30 days) cooling can be maintained. The level specified ensures enough net positive suction head (NPSH) available for operating the SSWS Pumps. The 24-hour frequency is adequate to verify level and for noticing trends.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.2

This SR verifies the SSWS can cool the ~~CCW System~~ CCWS to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a DBA. The 24-hour frequency is adequate to verify temperature and for noticing trends.

REFERENCES

1. CESSAR-DC Section 9.2.5, Ultimate Heat Sink.

2. Regulatory Guide 1.27 (Rev. 01), "Ultimate Heat Sink for Nuclear Power Plants."

3. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Standard Technical Specifications - Volume 1 (Criteria Application)."

Additional References

4. 10 CFR 50, Appendix A, GDC 5 - Sharing of Structures, Systems and Components.

5. 10 CFR 50, Appendix A, GDC 44 - Cooling Water.

6. NUREG-0800, "Standard Review Plan", Section 9.2.5, Rev. 2, "Ultimate Heat Sink," with attached Branch Technical Position ASB 9-2 "Residual Heat Energy for Light-Water Reactors for Long-Term Cooling."

16A.10.10 B 3.7.10 FUEL STORAGE POOL - WATER LEVEL

B 3.7 PLANT SYSTEMS

B 3.7.10 Fuel Storage Pool - Water Level

BASES

BACKGROUND

The minimum water level in the Fuel Storage Pool meets the assumptions of Iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are at their maximum capacity. The water also provides shielding during the movement of spent fuel. If normal cooling is lost, the water provides about a 12 hour heat sink before boiling occurs.

A general description of the Fuel Storage Pool design is found in CESSAR-DC Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in CESSAR-DC Section 15.7.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The minimum water level in the Fuel Storage Pool meets the assumptions of the fuel handling accident described in RG 1.25 (Ref. 3). The resultant two-hour thyroid dose to a person at the exclusion area boundary (EAB) is a small fraction of the 10 CFR 100 (Ref. 4) limits.

The assumption of RG 1.25, preserved by this specification, is that there is 23 feet of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel handling accident. With 23 feet, the assumptions of RG 1.25 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. However, in the case of a single bundle dropped and lying horizontally on top of the spent fuel racks there may be less than 23 feet above the top of the fuel bundle and the surface; by the width of the bundle. To offset this small non-conservatism, the analysis assumes that all 236 fuel rods fail; although analysis (Ref. 5) shows that only the first four rows, 60 fuel rods, fail from a hypothetical maximum drop. Also, due to the configuration of the fuel handling equipment, this type of drop is highly unlikely.

The Fuel Storage Pool Water level satisfies the requirements of Criterion 3 of the Interim Policy Statement as described in Reference 6.

(continued)

BASES

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 Fuel Storage Pool.

APPLICABILITY

This LCO applies whenever irradiated fuel is in the Spent Fuel Storage Pool because the potential for a release of fission products exists.

ACTIONS

A.1

When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. With the Fuel Storage Pool level less than required, the movement of spent fuel is brought to a halt in a safe position. This effectively precludes a spent fuel handling accident from occurring. The completion time is based on the operators capability to immediately move the fuel to a safe position when the problem becomes known. Plant procedures control the movement of loads over the spent fuel in such a case.

A.2.1 and A.2.2

Action to restore the water level should commence within a short period of time and be carried through to completion.

RA A.1, A.2.1 and A.2.2 are modified by a Note which allows an exemption to LCO 3.0.3. LCO 3.0.3 is not applicable as events in the Fuel Storage Pool are not affected by either MODE level or unit operations.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

This SR verifies sufficient water is available in the event of a fuel handling accident. The water level in the Fuel Storage Pool must be checked periodically. Because the ability to change water level is administratively controlled by plant procedures, a frequency of once-per-week is adequate.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

During refueling operations , the level in the fuel pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily under SR 3.9.6.1.

REFERENCES

1. CESSAR-DC Section 9.1.2, Spent Fuel Storage.

2. CESSAR-DC Section 15.7.4, Fuel Handling Accident.

3. Regulatory Guide 1.25 (Rev. 00), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

4. 10 CFR 100, Reactor Site Criteria.

5. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Standard Technical Specifications - Volume 1 (Criteria Application)."

Additional References

6. CESSAR-DC Section 9.1.3, Spent Fuel Pool Cooling and Cleanup System.

7. 10 CFR 50, Appendix A, GDC 61 - Fuel Storage and Handling and Radioactivity Control.

8. 10 CFR 50, Appendix A, GDC 64 - Monitoring of Radioactivity Releases.

9. Regulatory Guide 1.13 (Rev.01), "Spent Fuel Storage Facility Design Basis."

10. ANSI/ANS 2.19-1981, "Guidelines for Establishing Site Related Parameters for Site Selection and Design of an Independent Spent Fuel Storage Installation (Water Pool Type)."

16A.10.11 B 3.7.11 ATMOSPHERIC DUMP VALVES

B 3.7 PLANT SYSTEMS

B 3.7.11 Atmospheric Dump Valves

BASES

BACKGROUND	The Atmospheric Dump Valves (ADVs) provide a safety grade method for cooling the plant to Shutdown Cooling System (SCS) entry conditions should the preferred heat sink via the Steam Bypass System to the condenser not be available. This is done in conjunction with the Emergency Feedwater System providing cooling water from the Emergency Feedwater Storage Tank (EFWST). The ADVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Steam Bypass System. Four ADV lines are provided. Each ADV line consists of one ADV and an associated block valve. Two ADV lines per Steam Generator are required to meet single failure assumptions following an event rendering one Steam Generator unavailable for Reactor Coolant System (RCS) heat removal. The ADVs are provided with upstream block valves to permit their being tested at power and to provide an alternate means of isolation. The ADVs are equipped with electric motor operators with positioning circuits to permit control of the cooldown rate. The ADVs are OPERABLE with only a DC power source available. In addition, hand wheels are provided for local manual operation. A description of the ADVs is found in CESSAR-DC Section 10.3.2 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The design basis of the Atmospheric Dump Valves (ADVs) is established by the capability to cool the plant to shutdown cooling entry conditions at the design rate of 75 °F/hr using both Steam Generators, each with two ADVs. This design is adequate to cool the plant to SCS entry conditions with only one ADV and one Steam Generator utilizing the cooling water supply available in the EFWST.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

In the accident analysis presented in Chapters 6 and 15 of the CESSAR-DC, the ADVs are not assumed to be used until the operator takes action to cool down the plant. Prior to any operator action, the main steam safety valves (MSSVs) are used to maintain the Steam Generators pressure and temperature at the MSSVs setpoint. This is typically 30 minutes following initiation of an event. (This may be less for a Steam Generator Tube Rupture (SGTR) event.) The limiting events are those which render one Steam Generator unavailable for RCS heat removal, with a coincident loss of offsite power as a result of turbine trip and the single failure of one ADV on the unaffected Steam Generator. Typical initiating events falling into this category are a main steam line break (MSLB) upstream of the main steam isolation valves, a feedwater line break (FWLB), and a SGTR event (although the ADVs on the affected Steam Generator may still be available following a SGTR event).

The design must accommodate the single failure of one ADV to open on demand. Thus, each Steam Generator must have at least two ADVs, or the accident analysis must address cooldown for the above events.

The ADVs satisfy the requirements of Criterion 3 of the Interim Policy Statement as described in Reference 2.

LCO

Two ADVs are required on each Steam Generator to ensure that at least one ADV is available to conduct a plant cooldown following an event in which one Steam Generator becomes unavailable, accompanied by a single active failure of one ADV on the unaffected Steam Generator. The block valves must be OPERABLE to isolate an ADV. A closed block valve does not render it or its ADV inoperable if operator action time to open the block valve is supported in the accident analysis.

Failure to meet the LCO can result in the inability to cool the plant to SCS entry conditions following an event in which the condenser is unavailable for use with the Steam Bypass System.

An ADV is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow and is capable of fully opening and closing on demand.

(continued)

BASES

APPLICABILITY

In MODES 1, 2, and 3 the ADVs provide the safety grade path for cooling the RCS to SCS entry conditions following a SGTR.

In MODES 4, 5, and 6 a SGTR is not a credible event.

ACTIONS

A.1

With one ADV line inoperable, action should be taken to return the inoperable ADV line to OPERABLE status. The seven-day Completion Time is conservative as the ADV lines have a non-safety grade backup in the Steam Bypass System.

A Note has been added providing an exemption to LCO 3.0.4 for plants having more than one ADV per Steam Generator.

B.1

With more than one ADV line inoperable, action must be taken to restore at least three of the ADV lines to OPERABLE status. As the block valve can be closed to isolate an ADV, some repairs may be possible with the plant at power. The 24 hour Completion Time is reasonable based on the probability of an event occurring that requires the ADVs. The MSSVs, the Steam Bypass System, and any of the OPERABLE ADV lines are available to provide Steam Generator heat removal.

C.1 and C.2

If the ADVs cannot be restored to OPERABLE status in the associated Completion Time, the plant must be placed in a MODE in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 3 in 6 hours and in MODE 4 in 12 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODEs from full power operation without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies the OPERABILITY of the ADVs. To perform a controlled cooldown of the RCS, the ADVs must be able to be opened and throttled through their full range. This SR ensures the ADVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing, or use of an ADV during a plant cooldown may satisfy this requirement. This surveillance frequency is based on the length of a fuel cycle and has been shown to be adequate through operating experience.

SR 3.7.11.2

This SR verifies the OPERABILITY of the block valves. The function of the block valve is to isolate a failed open ADV. Cycling the block valve closed and open demonstrates its capability to perform this function. Performance of inservice testing, or use of the block valve during plant cooldown may satisfy this requirement. Although the actuation logic is tested as part of the SIAS functional test every 92 days, the subgroup relays that actuate the system cannot be tested during normal plant operation. The surveillance interval of 18 months is based on engineering judgment, and has been shown to be acceptable through operating experience.

REFERENCES

1. CESSAR-DC Section 10.3, Main Steam Supply System.

2. Thomas E. Murley (NRC) letter to Joseph K. Gasper (CEOG) dated May 9, 1988, forwarding the "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications."

Additional References

3. 10 CFR 50, Appendix A, GDC 34 - Residual Heat Removal.

4. 10 CFR 50, Appendix A, GDC 44 - Cooling Water.

5. Regulatory Guide 1.139, "Guidance for Residual Heat Removal," May 1978.

(continued)

BASES

REFERENCES

Additional References (continued)

6. ANSI/ANS 58.11-1983, "Cooldown Criteria for Light Water Reactors."

7. NUREG-0800, "Standard Review Plan", Section 10.3, Rev. 3, April 1984, "Main Steam Supply System."

8. Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat removal System" Rev. 2 - July 1981, attached to Section 5.4.7 "Residual Heat Removal", of NUREG-0800, "Standard Review Plan."

Complex

16A.10.12 B 3.7.12 CONTROL BUILDING VENTILATION SYSTEM

B 3.7 PLANT SYSTEMS
Cmlx

B 3.7.12 Control Building Ventilation System

BASES

BACKGROUND

The Control Building Ventilation and Air Conditioning Systems are designed to maintain the environment in the control room envelope and balance of control building within acceptable limits for the operation of unit controls, for maintenance and testing of the controls as required, and for uninterrupted safe occupancy of the control building area during post-accident shutdown. These systems are designed in accordance with the requirements of General Design Criteria 2, 4, 5, 19, and 60. Refer to Section ~~6.5~~ 6.4 for further information regarding control room habitability.

The main control room air-handling system is divisionally seperated and consists of two redundant air-handling units, each with filters, essential chilled water cooling coils for heat removal, and fans for air circulation. Some essential ducting is shared by both divisions for space consideration. The emergency circulation system consists of filter trains with particulate filters, carbon filters, and fans for emergency air circulation. Chilled water is supplied from the essential chilled water system.

During normal operation, return air from the control room is mixed with a small quantity of outside air for ventilation, is filtered and conditioned in the control room air-handling unit, and is delivered to the control room through supply ductwork. Duct-mounted heating coils and humidification equipment provide final adjustments to the control room temperature and humidity for maintaining normal comfort conditions.

In the event of ~~a loss-of-coolant-accident or~~ a release of toxic gases, the habitability zone is isolated from the outside environment~~, and the emergency circulation system is actuated to pressurize the control room.~~ In the event of a SIAS, both divisions of the main control room air-handling equipment ~~is~~ are automatically actuated ~~(or continues to operate, as applicable)~~ to remove heat and provide mixing~~and circulating~~, and filtration of the control room air. The emergency circulation system filters particulates and potential radioactive iodines from a portion of the return air, and delivers the filtered air to the inlet of the main air-handling unit.

(continued)

BASES

BACKGROUND (continued)

The Technical Support Center air-handling system consists of an air-handling unit, return air ~~and smoke purge~~ fans, and an emergency filter unit. The computer room air-handling system consists of two 100% air-handling units and associated fans. Both the Technical Support Center and computer room air-handling systems are non-safety and non-seismic.

The balance of control building air-handling systems consists of two redundant air-handling units, each with roughing filters, essential chilled water cooling coils and fans serving Division I electrical rooms, channel A and channel C. Two equal units are serving Division II channel B and D. Each Division will function with one of the redundant air handling units delivering filtered, conditioned air to the various electrical equipment rooms. Chilled water is supplied from the essential chilled water system. Each Division also contains redundant battery rooms with fans operating continuously to prevent buildup of hydrogen fumes. The safe shutdown area is served by Division II.

Return air from the various essential electrical equipment areas is mixed with a portion of outside air for ventilation, is filtered and conditioned in the air-handling unit, and is delivered to the rooms through supply ductwork. Duct-mounted heating coils provide final adjustments to temperature in selected equipment rooms.

The Operation Support Center, Men's Change, Women's Change, Break Room, Shift Assembly and Offices, Radiation Access Control and Cas. and Sec. Group areas all are served by an individual air handling unit consisting of a centrifugal fan, non essential chilled water coil and roughing filter.

The control room and technical support center
~~All of these areas can~~ receive outside air from the cleanest of two sources. ~~described for the control room.~~

The air entering the Control Room is continuously monitored by radiation and toxic gas detectors. One detector above the setpoint will cause actuation of the emergency radiation mode or toxic gas isolation mode as required. ~~The actions of the toxic gas isolation mode are more restrictive and will override the actions of the emergency radiation mode.~~

(continued)

BASES

BACKGROUND (continued)

A single division will pressurize the Control Room to ~~about~~ at least [0.125] inches water gauge, and provides an air ~~exchange~~ recirculation rate in excess of [25%] per hour. The Control ~~Building~~ Complex Ventilation Systems operation in maintaining the Control Room habitable is discussed in CESSAR-DC Section 6.4 (Ref. 1).

Redundant supply and recirculation division provide the required filtration should an excessive pressure drop develop across the other filter division. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. Redundant detectors for radiation and toxic gas protection are provided. The Control ~~Building~~ Complex Ventilation System is designed in accordance with Seismic Category I requirements.

The Control ~~Building~~ Complex Ventilation System is designed to maintain the Control Room environment for 30 days continuous occupancy after a DBA without exceeding 5 rem whole body dose.

APPLICABLE SAFETY ANALYSES

The Control ~~Building~~ Complex Ventilation System components are arranged in redundant safety-related ventilation divisions. The location of components and ducting within the Control Room envelope ensures an adequate supply of filtered air to all areas requiring access. During emergency operation the Control ~~Building~~ Complex Ventilation System maintains the temperature between 74°F and 85°F. The Control ~~Building~~ Complex Ventilation System provides airborne radiological protection for the Control Room operators as demonstrated by the Control Room accident dose analyses for the most limiting design basis Loss Of Coolant Accident fission product release presented in CESSAR-DC Chapter 15 (Ref. 2).

The analysis of toxic gas releases demonstrates that the toxicity limits are not exceeded in the Control Room following a toxic chemical release as presented in CESSAR-DC Section 6.4 (Ref. 1).

The balance of the control building air-handling system consists of two independent, full capacity systems. Each system serves the associated division of essential electrical equipment areas. Each system is powered from associated independent Class 1E power sources and served from ~~separate~~ essential chilled water systems. Equipment capacities are based on conservative evaluations of heat-producing equipment and conservative assumptions of

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

surrounding area temperatures. Normally, the electrical equipment areas will be maintained at approximately 85°F. The design basis upper limit of 104°F is based on standard ratings for electrical equipment.

All essential components of the control equipment area ventilation system are powered from Class 1E, diesel-backed power sources. Capacity of the control room air-handling system is based on complete failure of one division. Capacity and evaluation of the control room emergency circulation system is also based on complete failure of one division. The control room area air-handling systems are tied to the divisions of equipment that are served. Failure of one division of the control room area air-handling system may cause subsequent loss of components in the associated rooms. The consequences of this are acceptable since full redundancy of electrical components and electrical equipment areas is provided. System capacity is selected on the basis of a normal operating temperature of 85°F, or on a post-accident temperature of 104°F, whichever requires greater capacity.

A single active failure of a component of the Control Complex ~~Building~~ Ventilation System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The Control Complex ~~Building~~ Ventilation System is not in the primary success path for any accident analysis. In Reference 3, the NRC specified that the Control Complex ~~Building~~ Ventilation System met selection criterion 3 of the NRC Interim Policy Statement (Ref. 4) and that the LCO be retained in the technical specifications.

LCO

Two independent and redundant divisions of the Control Complex ~~Building~~ Ventilation System are required to ensure that at least one is available, assuming a single failure disables the other division. Total system failure could result in exceeding a dose of 5 rem to the Control Room operators in the event of a large radioactive release.

The Control Complex ~~Building~~ Ventilation System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both divisions. A division is considered OPERABLE when:

(continued)

BASES

LCO (continued)

1. its associated fan is OPERABLE, and
2. its associated HEPA filter and carbon adsorber are not excessively restricting flow and are capable of performing their filtration functions, and
3. its associated heater, demister, ductwork, valves and dampers are OPERABLE and air circulation can be maintained.

In addition, the Control Room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

APPLICABILITY

In MODES 1 and 2, the Control ~~Building~~ Complex Ventilation System must be OPERABLE to control operator exposure during and following a DBA.

In MODES 3 and 4, the Control ~~Building~~ Complex Ventilation System must be OPERABLE to cope with a hypothetical radiological release outside Containment which requires isolation of the Control Room.

In MODES 5 and 6, the Control Building Ventilation System may be required to cope with the release from the rupture of an outside waste gas tank.

During movement of irradiated fuel, the Control ~~Building~~ Complex Ventilation System must be OPERABLE to cope with the release from a fuel handling accident.

ACTIONS

A.1

With one Control ~~Building~~ Complex Ventilation System division inoperable, the inoperable Control ~~Building~~ Complex Ventilation System subsystem must be restored to OPERABLE status within seven days. In this condition, the remaining OPERABLE Control ~~Building~~ Complex Ventilation System subsystem is adequate to perform Control Room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced Control ~~Building~~ Complex Ventilation System capability. The seven-day Completion Time is based on the need for the Control ~~Building~~ Complex Ventilation

(continued)

BASES

ACTIONS (continued)

System functions and the low probability of a DBA occurring during this time period, and considering that the remaining subsystem can provide the required capabilities.

B.1 and B.2

In MODES 1, 2, 3, or 4, when RA A.1 cannot be completed within the required Completion Time, or if both Control ~~Building~~ Complex Ventilation System divisions are inoperable, the plant must be placed in a MODE which minimizes the accident risk. This is done by placing the plant in at least MODE 3 in 6 hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power operation without challenging plant systems.

C.1

In MODES 5 and 6, or during movement of irradiated fuel, when RA A.1 cannot be completed within the required Completion Time, the OPERABLE Control Building Ventilation System division should be immediately placed in the emergency radiation protection mode. ~~This action ensures that the remaining division is OPERABLE and that no failures which would prevent automatic actuation will occur and that any active failure will be readily detected.~~

RA C.1 is modified by a Note to alert the operator to place the system in the conservative mode of operation.

C.2.1, C.2.2, and C.2.3

An alternative to RA C.1 is to immediately suspend activities that present a potential for releasing radioactivity which might enter the Control Room. This places the plant in a condition which minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

The actions for Condition C are modified by a Note that LCO 3.0.3 is not applicable. This avoids the requirement to place the plant in cold shutdown if the actions for handling irradiated fuel are not met.

(continued)

BASES

ACTIONS (continued)

D.1, D.2, and D.3

~~When in MODES 5 and 6, and~~ During movement of irradiated fuel with two Control ~~Building~~ Complex Ventilation System divisions inoperable, the Required Action is to immediately suspend activities that present a potential for releasing radioactivity which might enter the Control Room. This places the plant in a condition which minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies that a division in a standby mode ~~starts~~ is available on demand and continues to operate. Standby systems should be checked periodically to ensure they start and function properly. As the environment and normal operating conditions on this system are not severe, testing each division once every month provides an adequate check on this system. Monthly heater operation dries out any moisture accumulated in the carbon from humidity in the ambient air. Systems without heaters need only be operated for 15 minutes to demonstrate the function of the system.

SR 3.7.12.2

Testing the performance of the HEPA filter, carbon adsorber efficiency, minimum system flow rate, and the physical properties of the activated carbon will be performed consistent with Regulatory Guide 1.52 requirements.

SR 3.7.12.3

This SR demonstrates that on an actual or simulated actuation signal each Control Building Ventilation System division starts and operates. The frequency of 18 months is specified in RG 1.52 (Ref. 5).

(continued)

Amendment I
December 21, 1990

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.4

This SR demonstrates the integrity of the Control Room enclosure and the assumed inleaking rates of the potentially contaminated air. The Control Room positive pressure with respect to potentially contaminated adjacent areas is periodically tested to verify proper function of the Control Building Ventilation System. During the emergency mode of operation, the Control Building Ventilation System is designed to pressurize the Control Room to [0.125] inches water gauge positive pressure with respect to adjacent areas to prevent unfiltered inleakage. The Control Building Ventilation System is designed to maintain this positive pressure at a flow rate of 2,000 cfm to the Control Room. The frequency of 18 months is consistent with the guidance provided in NUREG 0800, Section 6.4 (Ref. 7)

REFERENCES

1. CESSAR-DC Section 6.4, Habitability Systems.

2. CESSAR-DC Chapter 15, Accident Analysis.

3. CESSAR-DC Section 9.4.1, Control ~~Building~~ Complex Ventilation System.

4. Thomas E. Murley (NRC) letter to Joseph K. Gasper (CEOG) dated May 9, 1988, forwarding the "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications."

5. 52 FR 3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", February 6, 1987.

6. Regulatory Guide 1.52 (Rev. 02), "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."

7. NUREG-0800, "Standard Review Plan," Section 6.4, "Control Room Habitability System," Rev. 2, July 1981.

(continued)

BASES

REFERENCES (continued)

Additional References

8. 10 CFR 50, Appendix A, GDC 4 - Environmental and Missile Design Basis.

9. 10 CFR 50, Appendix A, GDC 19 - Control Room.

10. ANSI/ASME N509-1980, "Nuclear Power Plant Air Cleaning Units and Components." (Note: SRP 6.5.1, Rev. 2, July 1981 approved the use of the 1980 version, rather than the 1976 version in Ref. 5.).

11. ANSI/ASME N510-1980, "Testing of Nuclear Air Cleaning Systems." (Note: SRP 6.5.1, Rev. 2, July 1981 approved the use of the 1980 version, rather than the 1975 version in Ref. 9.)

12. NRC Generic Letter 83-13, "Clarification of Surveillance Requirements for HEPA Filters and Charcoal Absorber Units in Standard Technical Specifications on ESF Cleanup Systems," March 2, 1983.

13. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," June 1974.

14. Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," Rev. 1, January 1977.

VENTILATION SYSTEM (CRVS)

16A.10.13 B 3.7.13 CONTROL ROOM ~~EMERGENCY AIR TEMPERATURE CONTROL (HVAC) SYSTEM~~

B 3.7 PLANT SYSTEMS

B 3.7.13 Control Room ~~Emergency Air Temperature Control (HVAC) System~~
Ventilation System (CRVS)

BASES

BACKGROUND

The Control Room ~~Emergency Air Temperature Control (CREHVAC)~~ provides temperature control for the Control Room following isolation of the Control Room.

CRVS
The ~~CREHVAC~~ consists of two independent, redundant divisions which provide cooling and heating of recirculated Control Room air. Each division consists of a heating coils, cooling coils, instrumentation and controls to provide for Control Room temperature control. The ~~CREHVAC~~ CRVS is a subsystem providing air temperature control for the Control ~~Building~~ Complex Ventilation System, LCO 3.7.12.

CRVS
The ~~CREHVAC~~ is a standby system, parts of which may also operate during normal plant operations. A single train will provide the required temperature control to maintain the Control Room between ~~74°F and 85°F.~~ 73°F and 78 The CRVS ~~CREHVAC~~ operation in maintaining the Control Room temperature is discussed in CESSAR-DC Section ~~6.4~~ 9.4 (Ref. ~~1~~ 2).

APPLICABLE SAFETY ANALYSES

CRVS
The design basis of the ~~CREHVAC~~ is to maintain the Control Room environment habitability for 30 days continuous occupancy.

CRVS
The ~~CREHVAC~~ components are arranged in redundant safety related divisions. During emergency operation the ~~CREHVAC~~ CRVS maintains the temperature between ~~74°F and 85°F.~~ A single active failure of a component of the ~~CREHVAC~~ CRVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for Control Room temperature control. CRVS ~~CREHVAC~~ is designed in accordance with Seismic Category I requirements. ~~CREHVAC~~ CRVS is capable of removing sensible and latent heat loads from the Control Room which include consideration of equipment heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ~~CREHVAC~~ CRVS is not in the primary success path for any accident analysis. In Reference 2, the NRC specified that the ~~CREHVAC~~ CRVS met selection criterion 3 of the NRC Interim Policy Statement (Ref. 3) and that the LCO be retained in the technical specifications.

LCO

Two independent and redundant divisions of the ~~CREHVAC~~ CRVS are required to ensure that at least one is available, assuming a single failure disabling the other division. Total system failure could result in exceeding equipment operating temperature limits.

The ~~CREHVAC~~ CRVS is considered OPERABLE when the individual components that are necessary to maintain the Control Room temperature are OPERABLE in both divisions. These components include the cooling coils, and associated temperature control instrumentation. In addition the ~~CREACS~~ CRVS must be OPERABLE to the extent that air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6 and during movement of irradiated fuel, the ~~CREHVAC~~ CRVS must be OPERABLE to ensure that the Control Room temperature will not exceed equipment OPERABILITY requirements following isolation of the Control Room.

ACTIONS

A.1

With one ~~CREHVAC~~ CRVS division inoperable, the inoperable ~~CREHVAC~~ CRVS division must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE ~~CREHVAC~~ CRVS division is adequate to perform Control Room temperature control function. The 30 day Completion Time is based on engineering judgement of the risk from an event requiring the inoperable ~~CREHVAC~~ CRVS division, considering that the remaining division can provide the required capabilities and alternate safety or non-safety related cooling means are available.

(continued)

ACTIONS
(continued)

B.1 and B.2

In MODES 1, 2, 3 or 4, when RA A.1 cannot be completed within the required Completion Time, or if both ~~CREHVAC~~ CRVS divisions are inoperable, the plant must be placed in a MODE which minimizes the accident risk. This is done by placing the plant in at least MODE 3 in 6 hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power operation without challenging plant systems.

C.1

In MODES 5 and 6, or during movement of irradiated fuel, when RA A.1 cannot be completed within the required Completion Time, the OPERABLE CRVS ~~CREHVAC~~ division should be immediately placed in operation. This action ensures that the remaining train is OPERABLE, that no failures which would prevent automatic actuation will occur, and that any active failure will be readily detected.

C.2.1, C.2.2, and C.2.3

An alternative to RA C.1 is to immediately suspend activities that present a potential for releasing radioactivity which might require isolation of the Control Room. This places the plant in a condition which minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

D.1, D.2, and D.3

When in MODES 5 and 6 or during movement of irradiated fuel with two CRVS ~~CREHVAC~~ divisions inoperable, the Required Action is to immediately suspend activities that present a potential for releasing radioactivity which might require isolation of the Control Room. This places the plant in a condition which minimizes the accident risk.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies that the heat removal capability of the system is sufficient to meet design requirements. This SR is performed at a frequency of 18 months and consists of a combination of testing and calculations. An 18-month Frequency is appropriate as significant degradation of ~~the CREHVAC~~ CRVS is not expected over this time period.

REFERENCES

1. CESSAR-DC Section 6.4, Habitability Systems.
2. CESSAR-DC Section 9.4.1, Control ~~Building~~ Complex Ventilation System.
3. Thomas E. Murley (NRC) letter to Joseph K. Gasper (CEOG) dated May 9, 1988, forwarding the "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications."
4. 52 FR 3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", February 6, 1987.

Additional References

5. 10 CFR 50, Appendix A, GDC 4 - Environmental and Missile Design Basis.
6. 10 CFR 50, Appendix A, GDC 19 - Control Room.
7. NUREG-0800, "Standard Review Plan," Section 6.4, "Control Room Habitability System," Rev. 2, July 1981.

16A.10.14 B 3.7.14 SUBSPHERE BUILDING VENTILATION SYSTEM

B 3.7 PLANT SYSTEMS

B 3.7.14 Subsphere Building Ventilation System

BASES

BACKGROUND

Each division of the Subsphere Building Ventilation System consists of a supply air unit with two supply fans, prefilter, cooling and heating coils and dampers, an exhaust air unit with larger capacity, a full filter train and two exhaust fans with dampers.

The essential mechanical equipment room cooling units consist of a cooling coil with recirculation fan and dampers to remove heat generated within the space. A recirculation cooling unit is provided in addition to a once-through ventilation system because the served areas are potentially contaminated. Applicable areas are as follows:

A. Safeguard component areas including Safety Injection pump rooms, Shutdown Cooling pump rooms, Containment Spray pump rooms, Fuel Pool Heat-X rooms, Fuel Pool Cooling pump rooms, Penetration rooms, and associated piping and valve galleries.

The essential mechanical equipment room ventilation units consist of a once through ventilation cycle utilizing supply fans and exhaust fans to remove heat and maintain space temperature control. All rooms are considered clean areas, and exhausted by ventilation system. Applicable areas are as follows:

A. Motor-driven emergency feedwater pump rooms.

B. Steam-driven emergency feedwater pump rooms.

The essential mechanical equipment room cooling systems are designed to maintain the space temperatures below 100°F at times when the served equipment must operate. At least one train of essential mechanical equipment rooms is maintained below 100°F assuming a single failure of an active component concurrent with a loss of offsite power. Exhaust fans are powered from the diesel generators.

(continued)

BASES

BACKGROUND (continued)

The essential mechanical equipment room cooling systems perform the required safety function following a safe shutdown earthquake, and are able to withstand the effects of appropriate natural phenomena such as tornadoes, floods, and hurricanes (GDC 2).

The Subsphere Building essential HVAC System is designed to limit the offsite and control room dose following a LOCA or DBA within the guidelines of 10 CFR 100. Radiological consequences are discussed in Chapter 15.

The Subsphere Building, Ventilation Systems are separated according to Divisions with each ~~50%~~ 100% exhaust system containing a filter train and two fans.

one 100%

During normal operation of the general ventilation system, outside air is supplied by ~~two 50%~~ capacity supply units and two 50% capacity supply fans. The air is filtered and then conditioned as needed by the heating and cooling coils. The exhaust air is processed through ~~two 50%~~ capacity particulate filter system~~s~~ and is discharged to the unit vent by two 50% capacity exhaust fans. Supply and exhaust fans are electrically interlocked such that the building will always remain under a slight negative pressure. In the event of a loss-of-coolant-accident, the general ventilation equipment will continue to operate normally as long as offsite power is available. On LOOP, the exhaust fans will be powered from the Class 1E diesel generators. This maintains the subsphere pump rooms at a slight negative pressure to direct all releases through the exhaust filter train. Ducts to areas with non-essential cooling units will be isolated to enable proper operation of the emergency equipment.

ONE 100%

Normal operation of the essential mechanical equipment room cooling and ventilation units is with the equipment operating as required to maintain space temperatures. The cooling systems will operate based on heat load as indicated by room temperature. In the event of a LOCA or DBA, all units are automatically started and will operate at full capacity throughout the event. Individual room units will start when the equipment in the room starts.

(continued)

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BASES

APPLICABLE SAFETY ANALYSES

The essential mechanical equipment room cooling consist of two completely redundant, independent full-capacity systems. Division I cooling system serves Division I essential mechanical equipment rooms, and Division II cooling system serves Division II essential mechanical equipment rooms. Each train is powered from independent, Class 1E power sources. (Units with chilled water cooling coils are headered on separate essential chilled water cooling systems.) Equipment capacities are selected based on conservative evaluations of heat-producing equipment and conservative assumptions of adjacent area temperatures. Failure of one train may cause subsequent loss of components in the associated rooms. The consequences of this are acceptable since full redundancy of essential mechanical components is provided.

All essential components of the mechanical equipment room cooling systems are designed as Seismic Category I equipment, and will remain functional following a design basis earthquake. Intake and exhaust structures are protected from wind-generated or tornado-generated missiles.

Redundant components of the essential mechanical equipment room cooling systems are physically separated and protected from internally generated missiles. When subjected to pipe break effects, the components are not required to operate because the served mechanical equipment is located in the same space as the cooling components. Therefore, a pipe break in the same mechanical safety train is the only possible means of affecting the cooling system.

The Subsphere Building essential HVAC exhaust filter divisions are designed to limit the offsite and control room dose within the guidelines of 10 CFR 100 and SRP 6.5.1, respectively (References 4 and 6).

The Subsphere Building Ventilation System satisfies the requirements of Criterion 3 of the Interim Policy Statement as described in Reference 6.

(continued)

BASES

LCO

Two independent and redundant divisions of the Subsphere Building Ventilation System are required to ensure that at least one is available, assuming a single failure coincident with a loss of offsite power disabling the other train. Total system failure could result in the atmospheric releases from the SIS Pump Room exceeding the 10 CFR 100 limits in the event of a DBA.

The Subsphere Building Ventilation System is considered OPERABLE when the individual components necessary to maintain the SIS Pump Room filtration are OPERABLE in both divisions. A division is considered OPERABLE when:

1. its associated fan is OPERABLE, and

2. its associated HEPA filter and carbon absorber are not excessively restricting flow and are capable of performing their filtration functions, and

3. its associated heater, ductwork, valves and dampers are OPERABLE and air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3 and 4 the Subsphere Building Ventilation System is required to be OPERABLE consistent with the OPERABILITY requirements of the SIS.

In MODES 5 and 6, with reactor vessel level < [119 Ft - 0 inches], the Subsphere Building Ventilation System is ~~not~~ required to be OPERABLE ~~since the SIS is not required to be OPERABLE.~~ to support SI pump OPERABILITY requirements.

(continued)

BASES

ACTIONS

A.1

With one Subsphere Building Ventilation System division inoperable, the inoperable Subsphere Building Ventilation System division must be restored to OPERABLE status within seven days. In this condition, the OPERABLE Subsphere Building Ventilation System divisions is adequate to perform the SIS Pump Room air filtration. During this time, the remaining OPERABLE division is adequate to perform the SIS Pump Room Exhaust Air Cleanup function. However, the overall reliability is reduced where a single failure in the OPERABLE division could result in loss of function.

The seven-day Completion Time is appropriate because the risk contribution of the system is less than that for the SIS and this system is not a direct support system for the SIS.

B.1 and B.2

The plant must be placed in a MODE in which the requirement does not apply if the Subsphere Building Ventilation System cannot be restored to OPERABLE status in the associated Completion Time. This is done by placing the plant in at least MODE 3 in 6 hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power operation without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies that a division, having been shutdown when not required for operation, starts on demand and continues to operate. Standby systems should be checked periodically to ensure they start and function properly. As the environment and normal operating conditions on this system are not severe, testing each division once every month provides an adequate check on this system. Monthly heater operation dries out any moisture which may have accumulated in the carbon from humidity in the ambient air. Normal operation of the system during required modes satisfies this SR.

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.14.2

Testing the performance of the HEPA filter, ~~charcoal adsorber efficiency,~~ minimum system flow rate, and the physical properties of the activated charcoal will be performed consistent with Regulatory Guide 1.52 requirements.

SR 3.7.14.3

This SR demonstrates that on an actual or simulated actuation signal each Subsphere Building Ventilation System division starts and operates. The frequency of 18 months is consistent with that specified in RG 1.52.

SR 3.7.14.4

This SR demonstrates the integrity of the SIS Pump Room enclosure. The ability of the SIS Pump Room to maintain a negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the Subsphere Building Ventilation System. During the emergency mode of operation, the Subsphere Building Ventilation System is designed to maintain a slight negative pressure in the SIS Pump Room with respect to adjacent areas to prevent unfiltered leakage. The Subsphere Building Ventilation System is designed to maintain this negative pressure at a flow rate of ~~5,000~~ cfm from the ~~SIS Pump Room~~ sub-sphere. The frequency of 18 months is consistent with the guidance provided in NUREG 0800, Section 6.5.1 (Ref. 6).

13,200

The minimum system flow rate maintains a slight negative pressure in the SIS pump room area and provides sufficient air velocity to transport particulate contaminants, assuming only one filter train operating.

The number of filter elements is selected to limit the flow rate through any individual element to about 1,000 cfm. This may vary based on filter housing geometry. The maximum limit ensures that flow through, and pressure drop across, each filter element is not excessive.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~The number and depth of the adsorber elements ensures that, at the maximum flow rate, the residence time of the air stream in the charcoal bed achieves the desired adsorption rate. At least a 0.125 second residence time is necessary for an assumed 95% efficiency.~~

The filters have a certain pressure drop at the design flow rate when clean. The magnitude of the pressure drop indicates acceptable performance and is based on manufacturer's recommendations for the filter and adsorber elements at the design flow rate. An increase in pressure drop and/or decrease in flow indicate that the filter is being loaded or are indicative of other problems with the system.

This test is conducted with the tests for filter penetration and thus an 18-month frequency, consistent with that specified in RG 1.52, is used here also.

REFERENCES

1. CESSAR-DC Section 9.4.5, Subsphere Ventilation System.

2. CESSAR-DC Section 15.6.5, Loss of Coolant Accidents.

3. Regulatory Guide 1.52 (Rev. 02), "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."

4. 10 CFR 100, Reactor Site Criteria.

5. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Standard Technical Specifications - Volume 1 (Criteria Application)."

6. NUREG-0800, "Standard Review Plan", Section 6.5.1, Rev. 2, "ESF Atmosphere Cleanup Systems", Rev. 2, July 1981.

(continued)

BASES

REFERENCES (continued)

Additional References

7. NUREG-0800, "Standard Review Plan", Section 15.6.5, Rev. 2, Appendix B, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage from Engineered Safety Feature Components Outside Containment," Rev. 1.

8. 10 CFR 50, Appendix A, GDC 41 - Containment Atmosphere Cleanup.

9. 10 CFR 50, Appendix A, GDC 42 - Inspection of Containment Atmosphere Cleanup Systems.

10. 10 CFR 50, Appendix A, GDC 43 - Testing of Containment Atmosphere Cleanup Systems.

11. 10 CFR 50, Appendix A, GDC 61 - Fuel Storage and Handling and Radioactivity Control.

12. 10 CFR 50, Appendix A, GDC 64 - Monitoring Radioactivity Releases.

13. ANSI/ASME N509-1980, "Nuclear Power Plant Air Cleaning Units and Components." (Note: SRP 6.5.1, Rev. 2, July 1981 approved the use of the 1980 version, rather than the 1976 version in Reference 4.)

14. ANSI/ASME N510-1980, "Testing of Nuclear Air Cleaning Systems." (Note: SRP 6.5.1, Rev. 2, July 1981 approved the use of the 1980 version, rather than the 1975 version in Reference 16.)

15. NRC Generic Letter 83-13, "Clarification of Surveillance Requirements for HEPA Filters and Charcoal Absorber Units in Standard Technical Specifications on ESF Cleanup Systems", March 2, 1983.

16. CESSAR-DC, Section 15.B "Shutdown Risk Report"

16A.10.15 B 3.7.15 FUEL BUILDING VENTILATION EXHAUST SYSTEM

B 3.7 PLANT SYSTEMS

B 3.7.15 Fuel Building Ventilation Exhaust System

BASES

BACKGROUND

The Fuel Building Ventilation Exhaust System (FBVES) filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident. The FBVES, in conjunction with other, normally operating systems also provides environmental control of temperature and humidity in the Fuel Pool area.

The Fuel Building Ventilation Exhaust System consists of two 100% capacity filter trains. This portion of the Fuel Building Ventilation System is an engineered safety feature. The two filter trains receive separate emergency power.

Each of the filter trains consists of a moisture eliminator, prefilter, electric heater, absolute filter, carbon filter and post filter. It is equipped with a bypass section. The normal mode of operation for the filter trains is in the bypass position. Radiation detection is provided in the duct system header, upstream of the filter train inlet to monitor radioactivity. Upon indication of high radioactivity in the exhaust duct system, the bypass dampers will automatically close and the filter train inlet dampers will automatically open to direct air flow through the filter trains. Air from the Fuel Building Exhaust System is directed to the unit vent, where it is monitored before release to the atmosphere.

The 100% exhaust air system is manually set to the filtered mode during all fuel handling operations.

The FBVES is discussed in Section 9.4.2 of CESSAR-DC (Ref. 1). It may be used for normal, as well as post-accident atmospheric cleanup functions. The primary purpose of the heaters is to maintain the relative humidity to an acceptable level consistent with iodine removal efficiencies per RG 1.52 (Ref. 3).

(continued)

BASES

APPLICABLE SAFETY ANALYSES	The design basis of the FBVES is established by a fuel handling accident. The system is evaluated using the assumptions defined in RG 1.25 (Ref. 4). The analysis of the effects and consequences of a fuel handling accident are presented in CESSAR-DC Section 15.7.4 (Ref. 2). Two types of system failures are considered in the analysis: a complete loss of function, or excessive leakage. Either type of failure may result in a lower removal efficiency for any gaseous and particulate activity released during a fuel handling accident. The FBVES satisfies the requirements of Criterion 3 of the Interim Policy Statement as described in Reference 5.
LCO	Two independent and redundant divisions of the FBVES are required to ensure that at least one is available, assuming a single failure disabling the other division. Total system failure could result in the atmospheric releases from the Fuel Building exceeding the 10 CFR 100 limits in the event of a fuel handling accident. The FBVES is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both divisions. A division is considered OPERABLE when: 1. its associated fan is OPERABLE, and 2. its associated HEPA filter ~~and carbon adsorber~~ are[15] not excessively restricting flow and are capable of performing their filtration functions, and 3. its associated heater, moisture eliminator, ductwork, valves and dampers are OPERABLE and air circulation can be maintained.

BASES

APPLICABILITY	During movement of irradiated fuel in the fuel building, the FBVES is required to be OPERABLE to alleviate the consequences of a fuel handling accident.

(continued)

BASES

ACTIONS

A.1

With one FBVES division inoperable, the inoperable FBVES division must be restored to OPERABLE status within seven days. In this condition, the remaining OPERABLE FBVES division is adequate to perform Fuel Building air filtration function. The seven day Completion Time is based on engineering judgement of the risk from an event requiring the inoperable FBVES division, considering that the remaining division can provide the required capabilities.

B.1 and B.2

When RA A.1 cannot be completed within the required Completion Time during movement of irradiated fuel in the fuel building, the OPERABLE FBVES division should be started immediately or fuel movement suspended. This action ensures that the remaining division is OPERABLE and that no radiation detection instrumentation failures that will prevent system operation will occur and that any active failures will be readily detected. If the system is not placed in service, this action requires suspension of fuel movement which precludes a fuel handling accident. This does not preclude the movement of fuel to a safe position.

C.1

When two divisions of the FBVES are inoperable during movement of irradiated fuel in the fuel building, action should be taken to place the plant in a condition in which the LCO is not applicable. For this LCO, this involves immediately suspending movement of irradiated fuel in the fuel building. This does not preclude the movement of fuel to a safe position.

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR demonstrates that a division in a standby mode starts on demand and continues to operate. Standby systems should be checked periodically to ensure they start and function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Monthly heater operation drys out any moisture accumulated in the carbon from humidity in the ambient air.

SR 3.7.15.2

Testing the performance of the HEPA filter, ~~carbon adsorber efficiency,~~ minimum system flow rate, and the physical properties of the activated carbon will be performed consistent with Regulatory Guide 1.52 requirements.

SR 3.7.15.3

This SR demonstrates that on an actual or simulated actuation signal each FBVES division starts and operates. The frequency of 18 months is specified in RG 1.52 (Ref. 3).

SR 3.7.15.4

This SR demonstrates the integrity of the Fuel Building enclosure. The ability of the Fuel Building Room to maintain a negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FBVES. During the emergency mode of operation, the FBVES is designed to maintain a slight negative pressure in the Fuel Building with respect to adjacent areas to prevent unfiltered leakage. The FBVES is designed to maintain this negative pressure at a flow rate of 25,000 cfm from the Fuel Building.

The frequency of 18 months is consistent with the guidance provided in NUREG 0800, Section 6.4 (Ref. 6).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 7.15.5

Operating the bypass damper is necessary to ensure the system functions properly. The OPERABILITY of the bypass damper is verified if it can be opened. A frequency of 18 months is specified in RG 1.52 (Ref. 3)

REFERENCES

1. CESSAR-DC Section 9.4.2, Fuel Building Ventilation System.

2. CESSAR-DC Section 15.7.4, Fuel Handling Accident.

3. Regulatory Guide 1.52 (Rev. 02), "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."

4. Regulatory Guide 1.25, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors.

5. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Standard Technical Specifications - Volume 1 (Criteria Application)."

6. NUREG-0800, "Standard Review Plan", Section 15.6.5, Rev. 2, Appendix B, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage from Engineered Safety Feature Components Outside Containment", Rev. 1.

(continued)

BASES

REFERENCES (continued)

Additional References

7. 10 CFR 50, Appendix A, GDC 41 - Containment Atmosphere Cleanup.

8. 10 CFR 50, Appendix A, GDC 42 - Inspection of Containment Atmosphere Cleanup Systems.

9. 10 CFR 50, Appendix A, GDC 43 - Testing of Containment Atmosphere Cleanup Systems.

10. 10 CFR 50, Appendix A, GDC 61 - Fuel Storage and Handling and Radioactivity Control.

11. 10 CFR 50, Appendix A, GDC 64 - Monitoring of Radioactivity Releases.

12. ANSI/ASME N509-1980, "Nuclear Power Plant Air Cleaning Units and Components." (Note: SRP Rev. 2, July 1981 approved the use of the 1980 version, rather than the 1976 version in Ref. 4.)

13. ANSI/ASME N510-1980, "Testing of Nuclear Air Cleaning Systems." (Note: SRP 6.5.1, Rev. 2, July 1981 approved the use of the 1980 version, rather than the 1975 version in Ref. 15).

14. NRC Generic Letter 83-13, "Clarification of Surveillance Requirements for HEPA Filters and Charcoal Absorber Units in Standard Technical Specifications on ESF Cleanup Systems", March 2, 1983.

15. Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Rev.1, December 1975.

16A.10.16 B 3.7.16 DIESEL BUILDING VENTILATION SYSTEM

B 3.7 PLANT SYSTEMS

B 3.7.16 Diesel Building Ventilation System

BASES

BACKGROUND

Each Diesel Building Ventilation System is designed to maintain the diesel building temperature between 40°F minimum and ~~110°F~~ 120°F maximum when the diesel is not operating, and between 40°F minimum and ~~120°F~~ 122°F maximum when the diesel is operating.

Each diesel building ventilation system consists of supply and exhaust fans ~~with associated ductwork,~~ dampers, and controls for the diesel room. Heat energy from the diesel engine and other sources is absorbed by the ventilation supply air and discharged to the building exterior by the exhaust fans.

~~Isolation dampers are provided for the supply and exhaust ductwork to protect the diesel generator and associated equipment during a tornado or high winds.~~ Each (Emergency Diesel Building Ventilation) system is powered from ~~independent~~ the associated Class 1E power source~~s~~. Normal Diesel Building Ventilation System is non-safety class.

Each diesel generator building receives emergency ventilation air for cooling from two 50% capacity axial supply fans, each equipped with a weatherproof intake louver, ~~filter,~~ damper, and associated motor driver. Emergency Ventilation air is exhausted from each diesel room through two 50% capacity axial exhaust fans and dampers.

The diesel generator building ventilation system fans are automatically activated in response to building temperature. These automatic controls sequence the fans to meet required cooling demands.

When the diesel generator is shut down, the ventilation system can be manually activated if necessary to provide cooling for maintenance or testing access. A low room temperature setpoint will shut down all fans in order to limit the minimum room temperature to 40°F and prevent freezing. Unit heaters will be installed to hold the room temperature above 40°F.

(continued)

BASES

BACKGROUND (continued)

A missile barrier is provided over each air intake and exhaust louver to prevent the penetration of a missile into either diesel generator building. Intake and exhaust ducts are protected by appropriate security barriers.

Unit heaters are cycled as necessary to maintain a minimum temperature of 40°F for freeze prevention. Heat losses from equipment are conservatively estimated based on calculations and operating experience.

A single failure will not prevent the diesel generator building Emergency ventilation system from performing the intended heat removal function. ~~Each~~ ventilation system is powered by a Class 1E electrical system capable of being fed from the associated diesel generator.

Essential components of the diesel generator building ventilation system are designed to Seismic Category I requirements and will remain functional following a safe shutdown earthquake.

~~Diesel generator ventilation divisions are located in separate buildings above the respective diesel generators.~~ Each penetration into the building is provided with protection from external missiles. No high or moderate energy piping is located in the vicinity of the ventilation equipment or controls.

APPLICABLE SAFETY ANALYSES

The Diesel Generator Building Ventilation System is a support system required for operation of Diesel generators. Hence the Design Bases Accidents which credit the operation of Diesel generators (see Tech Spec Section 3.8) are also applicable to the Diesel Generator Building Ventilation System.

LCO

The LCO requires that each Diesel Generator Building Ventilation System is operable to ensure that the Diesel Generators can perform their safety functions when required.

For the Diesel Generator Building Ventilation System to be considered operable, the Diesel Generator Building temperature must be within design limits.

(continued)

BASES

APPLICABILITY

Since both Diesel Generators are required to be operable in MODES 1, 2, 3 and 4 the Ventilation System for each diesel generator building must also be operable in MODES 1, 2, 3 and 4. This same requirement for MODES 5 & 6 applies when the Diesel Generators are required to be OPERABLE.

ACTIONS

A.1

With the ventilation system inoperable, a time limit of 72 hours is provided to return the ventilation system to operable status. The 72 hours time limit is consistent with the time allowed to return an inoperable diesel generator to operable status.

A.2

Since the ventilation system supports the operation of each diesel generator, the diesel generator must be declared inoperable if the ventilation system is declared inoperable.

B.1 and B.2

The plant must be placed in a MODE in which the LCO does not apply if the Diesel Generator Building Ventilation System cannot be returned to operable status within the associated completion time. This is done by placing the plant in MODE 3 in 6 hours and in MODE 5 in 36 hours. The allowed Completion Times are based on operating experience, to reach the required plant conditions from full power without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

Verification that the Diesel Building Temperature is within the limits ensures that the building ventilation system is functioning properly and that the diesel generators can perform their safety functions when required. A 12 hour frequency ensures that potential problems will be quickly identified.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.16.2

Periodically, the supply and exhaust fans and their controls must be tested to verify proper performance. A frequency of 18 months is judged to be adequate and is consistent with the major surveillance testing performed on the diesel generators.

REFERENCES

1. CESSAR-DC Section 9.4.4, "Diesel Generator Building Ventilation System".

2. J. K. Gasper (CEOG) Letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987" CEN-335, C-E Owners Group Restructured Technical Specifications - Volume 1 (Criteria Application).

16A.10.17 B 3.7.17 ESSENTIAL CHILLED WATER SYSTEM

B 3.7 PLANT SYSTEMS

B 3.7.17 Essential Chilled Water System

BASES

BACKGROUND

The Essential Chilled Water System (ECWS) provides a heat sink for the removal of process and operating heat from selected safety related air handling systems during a transient or accident.

The ECWS is a closed loop system consisting of two independent divisions. Each 100% capacity division includes a chilled water refrigeration unit, heat exchanger, ~~compressor~~ expansion tank, two pumps, chemical addition tank, piping, valves, controls and instrumentation. An independent, 100% capacity chilled water refrigeration unit cools each division. The ~~ECW system~~ ECWS chilled water refrigeration unit is actuated on high ECWS temperature and supplies chilled water to ~~the~~ essential HVAC units ~~in Emergency Safety Features (ESF) equipment areas, such as the main Control Room, electrical equipment room and Subsphere Building area,~~ during a design basis event.

~~The flow path for the ECW system includes the closed loop of piping to all serviced equipment, and branch lines up to the first normally closed isolation valve.~~

During normal operation, the Normal Chilled Water System (NCWS) performs the cooling function of the ECWS through the ECWS heat exchanger with one of the ECWS pumps recirculating chilled water through the system. The NCWS is a non-safety ~~grade~~ related system. Additional information about the design and operation of the system~~, along with a list of components served~~ can be found in CESSAR-DC Section 9.2.9 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the ECWS is to remove the post-accident heat load from ESF spaces following a design basis accident with a loss of offsite power. Each division provides chilled water to the HVAC units at the design temperature of 45°F~~, and flow rate of 648 gpm.~~

(continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)	The maximum heat load in the ESF pump room area occurs following a loss of coolant accident (LOCA). Hot fluid from the IRWST is supplied to the Safety Injection (SI) and Containment Spray (CS) Pumps. This heat load to the area atmosphere must be removed ʰy the ECW system to ensure these systems remain OPERABLE. During a more normal cooldown, the Shutdown Cooling System (SCS) piping also provides a heat load in areas served the by ECWS. The ECWS satisfies the requirements of Criterion 3 of the Interim Policy Statement as described in Reference 2.
LCO	The requirements for two ECWS divisions provides the required redundancy ensuring the system functions to remove post-accident heat loads, assuming the worst single failure. A division is considered OPERABLE when: 1. it has an OPERABLE pump and associated compression tank, and 2. the associated piping, valves, chiller unit and instrumentation on the safety related flowpath are OPERABLE. The isolation of ECW to other components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the ECWS.
APPLICABILITY	In MODES 1, 2, 3 and 4 the ECWS is required to be OPERABLE when a LOCA or other accidents, would require ESF operation. In MODES 5 and 6, potential heat loads are smaller and the probability of accidents requiring the ECWS are low.

(continued)

BASES

ACTIONS

A.1

With one ECWS division inoperable, the inoperable ECWS division must be restored to OPERABLE status within seven days. In this condition, the OPERABLE ECWS division is adequate to perform the cooling function. The seven-day Completion Time is appropriate because of the high reliability of offsite power and availability of the normal Heating, Ventilation and Air Conditioning (HVAC) System.

This Action may be satisfied by examining logs or other information to determine if the valves are out-of-service for maintenance or other reasons. It does not mean to perform the SRs needed to demonstrate OPERABILITY of the valves.

With more than one division inoperable, LCO 3.0.3 directs the plant to a safe state.

B.1 and B.2

The plant must be placed in a MODE in which the requirement does not apply if the ECWS cannot be restored to OPERABLE status in the associated Completion Time. This is done by placing the plant in at least MODE 3 in 6 hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power operation without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

Verifying the correct alignment for manual, power operated, and automatic valves in the ECWS flowpath provides assurance that the proper flowpaths exist for ECWS operation. This SR does not apply to valves which are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves which cannot be inadvertently misaligned, such as check valves. The 31-day frequency is based on engineering judgement considering the importance of these valves and the low probability of their misalignment.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.17.2

This SR demonstrates proper automatic operation of the ECWS. The surveillance interval of 18 months is based on the refueling cycle and has been shown to be acceptable through operating experience.

REFERENCES

1. CESSAR-DC Section 9.2.9 Essential Chilled Water System.

2. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Standard Technical Specifications - Volume 1 (Criteria Application)."

Additional References

3. 10 CFR 50, Appendix A, GDC-4 - Environmental and Missile Design Basis.

4. 10 CFR 50, Appendix A, GDC-44 - Cooling Water.

5. ANSI/ANS 59.1-1979, "Safety Related Cooling Water Systems in Nuclear Power Plants."

16A.10.18 B 3.7.18 NUCLEAR ANNEX VENTILATION SYSTEM

B 3.7 PLANT SYSTEMS

B 3.7.18 Nuclear Annex Ventilation System

BASES

BACKGROUND

The Nuclear Annex Building general ventilation supply systems consist of two 50% capacity supply units and two 50% capacity supply fans. Supply units contain filters, heating coils, and chilled water cooling coils served from the non-essential chilled water system.

The Nuclear Annex Building general ventilation exhaust systems consist of two 50% capacity particulate filtration exhaust units, and two 50% capacity exhaust fans that discharge to the unit vent. or one 100% depending on the division,

The essential mechanical equipment room cooling units consist of chilled water cooling coil, direct-drive centrifugal recirculation fan, and dampers and controls to achieve the desired operation. The chilled water coils are served from the essential chilled water system.

The essential mechanical equipment room ventilation units contain intake filters, direct-drive centrifugal supply and exhaust fans, and dampers and controls to achieve the desired operation. There are heating and cooling coils to temper the outside air as required. Applicable areas are as follows:

Component cooling water system equipment areas including CCWS pump rooms, ~~CCWS heat exchanger rooms,~~ and essential chilled water system pump and chiller rooms.

The essential mechanical equipment room cooling systems are designed to maintain the space temperatures below 100°F at times when the served equipment must operate. At least one train of essential mechanical equipment rooms is maintained below 100°F assuming a single failure of an active component concurrent with a loss of offsite power. Exhaust fans are powered from the diesel generators.

(continued)

BASES

BACKGROUND (continued)

The Nuclear Annex Ventilation System is designed to provide ventilation and heat removal for personnel access to non-essential areas of the building. The design temperature range for the non-essential building areas is 60°F to 100°F.

Non-essential equipment in the Nuclear Annex in areas served by non-essential HVAC includes the normal chillers and pumps, CVCS tanks and pumps, hot machine and hot tool rooms, and the sample rooms.

During normal operation of the general ventilation system, outside air is supplied by two ~~50%~~ 100% capacity supply units and one or two ~~50%~~ 100% capacity supply fans. The air is filtered and then conditioned as needed by the heating and cooling coils. The exhaust air is processed through two 50% capacity particulate filter systems or one 100% capacity system and is discharged to the unit vent by two ~~50%~~ 100% capacity exhaust fans. Supply and exhaust fans are electrically interlocked such that the building will always remain under a slight negative pressure with respect to the environment to assure that all potentially radioactive releases are monitored prior to atmospheric discharge. In the event of a loss-of-coolant-accident, the general ventilation equipment will continue to operate normally as long as offsite power is available. Ducts to areas with essential cooling units will be isolated to enable proper operation of the emergency equipment.

Normal operation of the essential mechanical equipment room cooling and ventilation units is with the equipment operating as required to maintain space temperatures. The cooling systems will operate based on heat load as indicated by room temperature. Int he event of a LOCA or DBA, all units are started with the equipment being served, and will operate at full capacity throughout the event.

The essential mechanical equipment room cooling systems are protected from the effects of internally generated missiles, pipe break effects, and water spray (GDC 4).

The essential mechanical equipment room cooling systems perform the required safety function following a safe shutdown earthquake, and are able to withstand the effects of appropriate natural phenomena such as tornadoes, floods, and hurricanes (GDC 2).

(continued)

SAFETY ANALYSES

The essential mechanical equipment room cooling consist of two completely redundant, independent full-capacity systems. Division I cooling system serves Division I essential mechanical equipment rooms, and Division II cooling system serves Division II essential mechanical equipment rooms. Each train is powered from independent, Class 1E power sources. (Units with chilled water cooling coils are headered on separate essential chilled water cooling systems.) Equipment capacities are selected based on conservative evaluations of heat-producing equipment and conservative assumptions of adjacent area temperatures. Failure of one train may cause subsequent loss of components in the associated rooms. The consequences of this are acceptable since full redundance of essential mechanical components is provided.

All essential components of the mechanical equipment room cooling systems are designed as Seismic Category I equipment, and will remain functional following a design basis earthquake. Intake and exhaust structures are protected from wind-generated or tornado-generated missiles.

Redundant components of the essential mechanical equipment room cooling systems are physically separated and protected from internally generated missiles. When subjected to pipe break effects, the components are not required to operate because the served mechanical equipment is located in the same space as the cooling components. Therefore, a pipe break in the same mechanical safety train is the only possible means of affecting the cooling system.

The Nuclear Annex Ventilation System satisfies the requirements of Criterion 3 of the Interim Policy Statement as described in Reference 6.

LCO

Two independent and redundant divisions of the Nuclear Annex Ventilation System are required to ensure that at least one is available, assuming a single failure coincident with a loss of offsite power disabling the other train.

The Nuclear Annex Ventilation System is considered OPERABLE when the individual components necessary to maintain the Essential Mechanical Equipment Rooms ventilation units and cooling unit are operable in both divisions. A division is considered OPERABLE when:

(continued)

BASES

LCO (continued)

1. its associated fans are OPERABLE, and

2. its associated cooler has essential chill water flow through the cooling unit, and

3. its associated ventilation heater, ductwork, valves and dampers are OPERABLE and air circulation can be maintained.

APPLICABILITY

5and6

In MODES 1, 2, 3, ~~and~~ 4 the Nuclear Annex Ventilation System is required to be OPERABLE consistent with the OPERABILITY requirements of the SIS.

~~In MODES 5 and 6, the Nuclear Annex Ventilation System is not required to be OPERABLE since the SIS is not required to be OPERABLE.~~

~~as necessary to support the operability requirements of CCW and SIS consistent with the OPERABILITY~~

ACTIONS

A.1

With one Nuclear Annex Essential Mechanical room cooling and ventilation system division inoperable, the inoperable train must be restored to OPERABLE status within seven days. In this condition, the OPERABLE Subsphere Building Ventilation System trains is adequate. However, the overall reliability is reduced where a single failure in the OPERABLE train could result in loss of function.

The seven-day Completion Time is appropriate because the risk contribution of the system is less than that for the SIS.

B.1 and B.2

The plant must be placed in a MODE in which the requirement does not apply if the Nuclear Annex Essential Mechanical room cooling and ventilation system cannot be restored to OPERABLE status in the associated Completion Time. This is done by placing the plant in at least MODE 3 in 6 hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power operation without challenging plant systems.

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.18.1

This SR verifies that a division, having bees shutdown when not required for operation, starts on demand and continues to operate. Standby systems should be checked periodically to ensure they start and function properly. As the environment and normal operating conditions on this system ar not severe, testing each train once every month provides an adequate check on this system. Systems without heaters need only be operated for 15 minutes to demonstrate the function of the system. Normal operation of the system during required modes satisfies this SR.

SR 3.7.18.2

This SR demonstrates that on an actual or simulated actuation signal each Nuclear Annex Essential Mechanical Equipment room cooling and ventilation systems division recirculation units start and operate and that ducts to the general ventilation system isolate. The frequency of 18 months is consistent with that specified in RG 1.52.

REFERENCES

1. CESSAR-DC Section 9.4.9 Nuclear Annex Ventilation System.

2. CESSAR-DC Section 15.6.5, Loss of Coolant Accidents.

3. Regulatory Guide 1.52 (Rev. 0?), "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."

4. 10 CFR 100, Reactor Site Criteria.

5. J. K. Gasper (CEOG) letter CEOG-87-735 to Dr. T. E. Murley (Director NRR/NRC) dated December 11, 1987 "CEN-355, C-E Owners Group Restructured Standard Technical Specifications - Volume 1 (Criteria Application)."

6. NUREG-0800, "Standard Review Plan", Section 6.5.1, Rev. 2, "ESF Atmosphere Cleanup Systems", Rev. 2, July 1981.

(continued)

BASES

REFERENCES (continued)

Additional References

7. NUREG-0800, "Standard Review Plan", Section 15.6.5, Rev. 2, Appendix B, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage from Engineered Safety Feature Components Outside Containment," Rev. 1.

8. 10 CFR 50, Appendix A, GDC-41 - Containment Atmosphere Cleanup."

9. 10 CFR 50, Appendix A, GDC 64 - Monitoring Radioactivity Releases.

10. ANSI/ASME N509-1980, "Nuclear Power Plant Air Cleaning Units and components." (Note: SRP 6.5.1, Rev. 2, July 1981 approved the use of the 1980 version, rather than the 1976 version in Reference 4.)

11. ANSI/ASME N510-1980, "Testing of Nuclear Air Cleaning Systems." (Note: SRP 6.5.1, Rev. 2, July 1981 approved the use of the 1980 version, rather than the 1975 version in Reference 16.)

B 3.7 PLANT SYSTEMS

B 3.7.~~17~~ 19 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND	As described in LCO 3.7.~~18~~ 20, "Spent Fuel Assembly Storage," fuel assemblies are stored in the spent fuel racks [in a "checkerboard" pattern] in accordance with criteria based on [initial enrichment and discharge burnup]. Although the water in the spent fuel pool is normally borated to 4000 ≥ [~~1800~~] ppm, the criteria which limits the storage of a fuel assembly to specific rack locations is conservatively developed without taking credit for boron.
APPLICABLE SAFETY ANALYSES	A fuel assembly could be inadvertently loaded into a spent fuel rack location not allowed by LCO 3.7.~~18~~ 20 (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). This accident is analyzed assuming the extreme case of completely loading the fuel pool racks with unirradiated assemblies of maximum enrichment. Another type of postulated accident is associated with a fuel assembly which is dropped onto the fully loaded fuel pool storage rack. Either incident could have a positive reactivity effect, decreasing the margin to criticality. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. The concentration of dissolved boron in the fuel pool satisfies Criterion 2 of the NRC Policy Statement.
LCO	The specified concentration of dissolved boron in the fuel pool preserves the assumptions used in the analyses of the potential accident scenarios described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel pool.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool until a complete spent fuel pool verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

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

The Required Actions are modified by a NOTE indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement fuel of assemblies. This does not preclude the movement of fuel assemblies to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~17~~ 19.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.

Leave in

~~REFERENCES None.~~

B 3.7 PLANT SYSTEMS

B 3.7.~~18~~ 20 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store [~~735~~] 907 irradiated fuel assemblies, which includes storage for [15] failed fuel containers. The spent fuel storage cells are installed in parallel rows with center to center spacing of 9.780 [~~12 31/32~~] inches in one direction, and [~~13 3/16~~] 9.780 inches in the other orthogonal direction. This spacing and "flux trap" construction~~, whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans,~~ using non-poisoned "L" inserts is sufficient to maintain a k_{eff} of $\leq$ 0.95 for spent fuel of original enrichment of up to [~~3.3%~~] 5%. However, as higher initial enrichment fuel assemblies are stored in the spent fuel pool, they must be stored in a checkerboard pattern taking into account fuel burn up to maintain a k_{eff} of 0.95 or less.

APPLICABLE SAFETY ANALYSES

The spent fuel storage facility is designed for noncriticality by use of adequate spacing, and "flux trap" construction~~, whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans.~~ using non-poisoned "L" inserts

The spent fuel assembly storage satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, according to [Figure 3.7.~~18~~ 20-1], in the accompanying LCO, ensures that the k_{eff} of the spent fuel pool will always remain < 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool according to [Figure 3.7.~~18~~ 20-1], in the accompanying LCO.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in [Region 2] of the spent fuel pool.

ACTIONS

A.1

Required Action A.1 is modified by a NOTE indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in [Region 2] the spent fuel pool is not in accordance with Figure [3.7.~~18~~ 20-1], immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure [3.7.~~18~~ 20-1].

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~18~~ 20.1

This SR verifies by administrative means that the initial enrichment and [discharge fuel] burnup of the fuel assembly is in accordance with Figure [3.7.~~18~~ 20-1] in the accompanying LCO.

REFERENCES

None.

16A.11 B 3.8 ELECTRICAL POWER SYSTEMS

16A.11.1 B 3.8.1 AC SOURCES - OPERATING

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

BACKGROUND

The AC Power Sources consist of the offsite power sources (preferred power) and the onsite standby power sources. As required by General Design Criterion 17 (Ref. 1), the design of the AC Power System provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The Division 1 and 2 onsite Class 1E AC Distribution System is divided into redundant load groups so that loss of any one group will not prevent the minimum safety functions from being performed. Each set of redundant AC load groups has connections to two preferred (offsite) power supplies and to a single diesel generator.

Independant ~~T~~transmission lines ~~from two separate transmission systems~~ supply offsite power to ~~the 230 kV switchyard. From the switchyard, two separate lines feed the Unit.~~ Preferred Switchyards I & II. Preferred Switchyard I ~~One line~~ feeds the Unit Main Transformers (UMTs) and Preferred Switchyard II ~~the other~~ feeds the Reserve ~~Standby~~ Auxiliary Transformers (RATs). The UMT transforms 230 kV to 24 kV through the four UMTs. This 24 kV is fed to two Unit Auxiliary Transformers (UAT). These UATs each provide power to their respective separate switchgear groups [x] and [y]. The UATs provide 13.8 kV for large motors and ~~11.6~~ 4.16 kV for primary station distribution.

UATs provide the normal preferred source of power to the 4160 volt emergency busses. UAT[x] provides the power to Division 1 emergency busses and UAT[y] provides the power to Division 2 emergency busses. Backup offsite power for either or both the emergency busses is provided through the ~~S~~RATs. If offsite power is not available, the emergency busses are supplied from their respective diesel generator, (DG). DG1 supplies power to Division 1 emergency busses and DG2 supplies power to Division 2 emergency busses.

If power were lost from either UAT, undervoltage relays would sense this condition. The electrical system would then ~~try~~ attempt to transfer to the backup preferred power source (the ~~SAT) and start the associated DG~~ associated RAT). If power is not available from the backup preferred source, the DG is automatically used to

The ~~fast~~ transfer to the associated RAT will occur on the permanent non-safety bus affected. A successful ~~fast~~ transfer will preclude the automatic start of the associated diesel generator.

(continued)

BASES

BACKGROUND (continued)

power the associated emergency busses. The DGs start automatically on a Safety Injection Actuation Signal (SIAS) or on a loss of voltage (LOV) on the respective emergency busses. Even though the DGs are started on SIAS, they will not power the emergency busses unless both preferred offsite sources of power are unavailable. The DG automatically ties to it's busses on a LOV condition on that bus without offsite power ~~available.~~ unavailable.

~~Loads are sequentially connected to their respective emergency busses by their automatic load sequencer. The load sequencer controls the start signals to motor breakers to prevent overloading the power source from starting all loads simultaneously. Certain critical loads such as diesel support loads are powered from transformers wired directly to the diesel generator but are still sequenced on the load sequencer. All required loads are powered via the load sequencer within 40 seconds after the initiation of the event.~~

In accordance with Regulatory Guide 1.9 (Ref. 2), diesel generators 1 and 2 have ~~[4700]~~ ~~5957~~ 6023 kW continuous and ~~[5170]~~ ~~6585~~ 6626 kW two-hour load ratings.

The diesel generators are rated at 4160 volts, three phase, 60 Hz, and are capable of attaining rated frequency and voltage within twenty seconds after receipt of a start signal (Ref. 3).

The ESF systems which are powered from divisional power sources are listed in Reference 4.

APPLICABLE SAFETY ANALYSES

The initial conditions of design basis transient and accident analyses in FSAR Chapters 6, Engineered Safety Features, and 15, Accident Analyses, assume ESF systems are OPERABLE. The AC Power System is designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These design limits are discussed in more detail in the Bases for LCO Sections 3.2 (Power Distribution Limits), 3.4 (Reactor Coolant System), and 3.6 (Containment Systems).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

In general, the safety analysis considered offsite power to be available to ESF equipment following event initiation. Offsite power is not considered to be safety-related. A loss of offsite power (LOOP) alone is an analyzed event since it presents a challenge to the plant's safety features and would result in a total loss of AC power if the diesel generators failed to start.

The OPERABILITY of an offsite AC source is not explicitly required by the safety analyses. Therefore, the need for two independent offsite power circuits was not derived from the safety analysis, since events postulating failure of offsite power considered a complete loss of 230kv power. Such events disable both offsite circuits. The requirement for two offsite circuits was derived from the design criteria (Ref. 1) and standards incorporated into the plant design, which required redundant, independent offsite power sources.

The OPERABILITY of the power sources is consistent with the initial assumptions of the accident analyses and design requirements and is based upon maintaining at least one of the AC and DC Power Sources and associated distribution systems OPERABLE during accident conditions in the event of (1) an assumed loss of all offsite or all onsite AC power, and (2) a single failure of the other AC source.

LCO

Two physically independent circuits (Ref. ~~6~~ 7) between the offsite transmission network and the onsite Class 1E Distribution System, and the two independent diesel generators (Ref. ~~7~~ 6) ensure availability of the required power to shutdown the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence or a postulated design basis accident (DBA).

The two circuits from offsite are required to be "physically independent" such that a single component fault (e.g., breaker trip) will not cause both power sources to be lost to one or more 4160 volt emergency buses. Thus, a physically independent circuit consists of one incoming line to the 230 kV ~~switchyard~~ (Preferred Switchyard Interface I), a circuit path (including breakers and disconnects) to one energized UAT (~~1~~ or ~~2~~), and a circuit path from the energized UAT to the associated 4160 volt emergency buses. A physically independant circuit also consists of the incoming line to the Preferred Switchyard Interface II, a circuit path (including breakers and disconnects) to one energized RAT (Division I or II), and a circuit path from that energized RAT to its 4160 volt emergency buses.

(continued)

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BASES

LCO (continued)

Inoperable AC sources do not necessarily result in inoperable components (which are designed to receive power from that source) unless specifically directed by Required Actions (refer to LCO 3.0.7).

Certain diesel generator support systems are addressed in other LCOs. During inoperabilities in these support systems, inoperable diesel generators do not necessarily result unless specifically directed by Required Actions. This is in accordance with LCO 3.0.7.

APPLICABILITY

The AC Power Sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

1. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of anticipated operational occurrences or abnormal transients, and

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

AC Power Source requirements for MODES 5 and 6 are addressed in LCO 3.8.2, AC Sources - Shutdown.

ACTIONS

Operating experience indicates that the availability of a typical offsite source is higher than that of a typical standby AC supply (Ref. 8). Thus, if risk is evaluated in terms of availability, the risk associated with the loss of an offsite power source (the source with the higher availability) would appear to be more severe than the risk associated with the loss of a standby AC supply (the source with the lower availability). However, this apparent difference in severity is offset by maintainability considerations: that is, the time required to detect and restore an unavailable offsite source is generally much less than that required to detect and restore an unavailable standby AC supply. Based on these considerations, a general distinction between operating restrictions associated with the loss of an offsite source and those restrictions associated with the loss of a standby AC supply is not warranted.

(continued)

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BASES

ACTIONS (continued)

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

With one of the required offsite circuits inoperable, sufficient offsite power is available from the other required offsite circuit to ensure that the unit can be maintained in a safe shutdown condition following a design basis transient or accident. Even failure of the remaining required offsite circuit will not jeopardize a safe shutdown of the unit because of the redundant standby diesel generator. However, since system reliability is degraded below the LCO requirements, a time limit on continued operation is imposed. To ensure a highly reliable power source remains, it is necessary to verify the availability of the remaining required offsite circuit on a more frequent basis when one offsite circuit is inoperable.

The specific list of features encompassed by Required Action A.2 is provided in Reference 8. These features are those which are designed with redundant safety-related divisions. Single division systems are not included. Since the Completion Time allowance for this Required Action is limited to 24 hours, those systems with allowed Completion Times ≥ to 24 hours for both divisions inoperable are not included as required features to be checked. 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 diesel generator will not result in a complete loss of safety function of critical systems. The term "ensure," as used in Required Action A.2, allows for an administrative check by examining logs or other information, to determine if certain features are out of service for maintenance or other reasons. It does not require unique performance of the Surveillance Requirements needed to demonstrate OPERABILITY of the feature. The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock". In this Required Action, the Completion Time only begins on discovery that both 1) the division has no offsite power supplying its loads, and 2) a required feature on the other division is inoperable. If at anytime during the existence of this Condition (one offsite circuit inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked. The Completion Time is based on engineering judgment taking into consideration the probability of an event concurrent with a single failure of the associated diesel generator, thereby leading to a loss of function of this feature.

(continued)

ACTIONS (continued)

A.1, A.2, and A.3 (continued)

Due to the loss of redundancy in the offsite power sources, a single failure in the offsite AC Power System could result in a complete loss of offsite power. Therefore, Required Action A.3 restricts continued operation. In accordance with Regulatory Guide 1.93 (Ref. 8), operation may continue for a period that should not exceed 72 hours. If the source is not restored within 72 hours, a controlled shutdown must be initiated per Required Actions F.1 and F.2.

B.1, B.2, B.3.1, B.3.2, and B.4

With one diesel generator inoperable, sufficient AC Power Sources remain available to ensure safe shutdown of the unit in the event of a transient or accident without a single failure. Operation could therefore safely continue for a short period of time if the availability of the remaining sources is verified.

The specific list of features encompassed by Required Action B.2 is provided in Reference 8. These features are those which are designed with redundant safety related divisions. Single division systems are not included. Since the Completion Time allowance for this Required Action is limited to four hours, those systems with allowed Completion Times ≥ four hours for both divisions inoperable are not included as required features to be checked. Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a diesel generator is inoperable, will not result in a complete loss of safety function of critical systems. The term "ensure," as used in Required Action B.2, allows for an administrative check by examining logs or other information, to determine if certain features are out of service for maintenance or other reasons. It does not require unique performance of the Surveillance Requirements needed to demonstrate OPERABILITY of the feature. The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both 1) an inoperable diesel generator exists, and 2) a required feature on the other division is inoperable. If at any time during the existence of this Condition (one diesel generator inoperable) a required feature subsequently becomes inoperable, this

(continued)

ACTIONS (continued)

B.1, B.2, B.3.1, B.3.2, and B.4 (continued)

Completion Time would begin to be tracked. The Completion Time is based on engineering judgment taking into consideration the probability of a loss of offsite power occurring while the other Division 1 or 2 diesel generator is inoperable. This is comparable to, but less severe than, Condition D (both diesel generators inoperable) and therefore has a comparable, but less restrictive, Completion Time.

Required Action B.3.1 provides an allowance to avoid unnecessary testing of the OPERABLE diesel generator. If it can be determined that the cause of the inoperable diesel generator does not exist on the OPERABLE diesel generator, SR 3.8.1.2 (diesel generator start) does not have to be performed. If the cause of inoperability exists on the other diesel generator, the second diesel generator would be declared inoperable upon discovery of the condition and Condition D would be entered. Once the failure is repaired and the common mode failure no longer exists, Required Action B.3.1 is satisfied. If the cause of the initial inoperable diesel generator cannot be confirmed to not exist on the remaining diesel generator, performance of SR 3.8.1.2 will suffice to provide assurance of continued OPERABILITY of that diesel generator.

Per Generic Letter 84-15 (Ref. 9), eight hours is a reasonable time to confirm that the OPERABLE diesel generator is not affected by the same problem as the inoperable diesel generator.

Required Action B.4 restricts continued operation to 72 hours in accordance with Regulatory Guide 1.93 (Ref. 8). If the inoperable diesel generator is not restored within 72 hours, a controlled shutdown must be initiated per Required Actions F.1 and F.2.

Condition B is modified by a Note which requires performance of Required Action B.3.1 or B.3.2 if Condition B is entered. Because of the potential for common cause failures, Required Action B.3.1 or B.3.2 must be completed when one diesel generator becomes inoperable even if it is restored to OPERABLE status in less than eight hours.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

In Condition C, individual redundancy is lost in both the offsite power system and the onsite Division 1 or 2 AC Power System. However, since power system redundancy is provided by two diverse sources of power, the reliability of the power systems in this Condition may appear higher than Condition E (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure.

During the time this Condition exists (one offsite circuit and one diesel generator inoperable), Condition A and B also exist concurrently. The Required Actions and associated Completion Times for these Conditions also apply from time of entry into each individual Condition. This will continue to provide common mode failure considerations for the inoperable diesel generator, cross-divisional feature OPERABILITY considerations, and provide the appropriate time limit for continued operation while repairs are being attempted.

Per Regulatory Guide 1.93 (Ref. 8), with the available offsite and standby AC Power Sources each one less than the LCO, operation may continue for 12 hours. If either an offsite or a standby AC source is restored to OPERABLE status within 12 hours, operation may continue for 72 hours from the time of the initial loss of the remaining inoperable source (consistent with the loss of one AC source in Condition A or B). If neither an offsite source nor a standby source is restored within the 12 hours, or, if either the inoperable diesel generator or the inoperable circuit is not restored within 72 hours of its initial inoperability in accordance with Condition A or B (which may occur, in some cases, prior to the 12-hour allowance), a controlled shutdown must be initiated per Required Actions F.1 and F.2.

(continued)

BASES

ACTIONS (continued)

D.1

With two required diesel generators inoperable, insufficient standby AC Power Sources are available to power the minimum required ESF functions. Since the offsite power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (i.e, the immediate shutdown could cause grid instability which could result in total loss of AC power). However, since any inadvertent generator trip could also result in total loss of AC power, the time allowed for continued operation is severely restricted. The intent here is not only to avoid the risk associated with an immediate controlled shutdown but also to minimize the risk associated with this level of degradation. During the time this condition exists (both diesel generators inoperable), Condition B also exists concurrently for each of the inoperable diesel generators independently. The Required Actions and associated Completion Times apply as discussed previously. This will continue to provide common mode failure considerations, cross-divisional feature OPERABILITY, and provide the appropriate time limit for continued operation while repairs are being attempted.

Per Regulatory Guide 1.93 (Ref. 8), with the available standby AC electrical supplies two less than the LCO, operation may continue for a period that should not exceed two hours. One of the required diesel generators must be restored within these two hours. Operation may then continue in accordance with the loss of one diesel generator in Condition B. If no standby AC supply is restored within two hours, or, if either inoperable diesel generator is not restored within 72 hours of its initial inoperability in accordance with Condition B (which may occur, in some cases, prior to the two-hour allowance), a controlled shutdown must be initiated per Required Actions F.1 and F.2.

E.1 and E.2

With both of the required offsite circuits inoperable, sufficient standby AC Power Sources are available to maintain the unit in a safe shutdown condition in the event of a design basis transient or accident. However, since AC Power System reliability is degraded below the LCO requirements, a time limit on continued operation is imposed.

(continued)

ACTIONS (continued)

E.1 and E.2 (continued)

The specific list of features encompassed by Required Action E.1 is provided in Reference 8. These features are those which are designed with redundant safety-related divisions. Single division systems are not included. Since the Completion Time allowance for this Required Action is limited to 12 hours, those systems with allowed Completion Times ≥ 12 hours for both divisions inoperable are not included as required features to be checked. The requirement is intended to provide assurance should a coincident single failure of a diesel generator occur during the period with two offsite circuits inoperable, a complete loss of safety function of critical systems will not result. The term "ensure," as used in Required Action E.1, allows for an administrative check by examining logs or other information, to determine if certain features are out of service for maintenance or other reasons. It does not require unique performance of the Surveillance Requirements needed to demonstrate OPERABILITY of the feature. The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock". In this Required Action the Completion Time only begins on discovery that 1) both offsite circuits are inoperable, and 2) a required feature on the other division is inoperable. If at any time during the existence of this Condition (both offsite circuits inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked. The Completion Time is based on engineering judgment taking into consideration the probability of an event concurrent with a single failure of a diesel generator occurring (on the division opposite to the inoperable feature) while two offsite circuits are inoperable. During the time this Condition exists (both offsite circuits inoperable), Condition A also exists concurrently for each of the inoperable offsite circuits independently. The Required Actions and associated Completion Times apply as discussed previously. This may result in more restrictive requirements for restoration and/or cross-divisional feature OPERABILITY checks.

In accordance with Regulatory Guide 1.93 (Ref. 8), with the available offsite AC Power Sources two less than required by the LCO, operation may continue for 24 hours. One offsite source must be restored within 24 hours. Operation may then continue in accordance with the loss of one offsite source in Condition A. If no offsite circuit is restored within 24 hours, or, if either

(continued)

BASES

ACTIONS (continued)

E.1 and E.2 (continued)

inoperable offsite circuit is not restored within 72 hours of its initial inoperability in accordance with Condition A (which may occur, in some cases, prior to the 24-hour allowance), a controlled shutdown must be initiated per Required Actions F.1 and F.2.

F.1 and F.2

The plant must be placed in a MODE in which the LCO does not apply if the Required Actions and associated Completion Times cannot be met. This is done by placing the plant in at least MODE 3 in six hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power without challenging plant systems.

G.1

With three or more required AC sources inoperable, insufficient AC sources remain available to ensure safe shutdown of the unit in the event of a transient or accident with any additional single failure. This Required Action precludes allowing operation to continue in combinations of Conditions A through E with three of four sources inoperable. *Immediately* is used as an administrative means of not allowing any extension of the LCO 3.0.3 shutdown requirements.

SURVEILLANCE REQUIREMENTS

The AC Power Sources are designed to permit inspection and testing of all important areas and features, especially those which have a standby function, in accordance with General Design Criteria 18 (Ref. 10). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guide 1.9 (Ref. 2), 1.108 (Ref. 11), and 1.137 (Ref. 12).

(continued)

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.1

This Surveillance Requirement assures proper circuit continuity for the offsite AC power supply to distribution network and availability of offsite AC power. The breaker alignment verifies that each breaker is in its correct position to ensure distribution buses and loads are connected to their preferred power source and independence of offsite circuits is maintained. The seven-day Frequency is adequate since breaker position is not likely to change without the operator being aware of it.

SR 3.8.1.2 and SR 3.8.1.5

These surveillances help to ensure the availability of the standby power supply to mitigate design basis transients and accidents and maintain the unit in safe shutdown conditions. For the purpose of this testing, the diesel generators shall be started from standby conditions. Standby conditions in this case means the diesel engine coolant and oil are being continuously circulated and temperature maintained consistent with manufacturer recommendations.

SR 3.8.1.5 requires, on a 184-day Frequency, the diesel generators start from standby conditions and achieve required voltage and frequency within 20 seconds. The 20-second requirement supports the assumptions in the design basis loss of coolant accident (LOCA) analysis (Ref. 3). The twenty-second start requirement is not applicable to SR 3.8.1.2 which is performed on a 31-day Frequency.

The normal 31-day Frequency for SR 3.8.1.2 (see Diesel Generator Test Schedule) is consistent with Regulatory Guide 1.108 (Ref. 11). The 184-day Frequency for SR 3.8.1.5 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 9). These Frequencies provide adequate assurance of diesel generator OPERABILITY while minimizing degradation resulting from testing.

The Diesel Generator is qualified to be adjustable for variable speed starting in accordance with manufacturer's recommendations for slow starts. The governor variable speed feature will be automatically bypassed on emergency starts allowing the Diesel Generator to start and reach rated speed and voltage within the required 20 seconds.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.2 and SR 3.8.1.5 (continued)

Several Notes modify the performance of SRs 3.8.1.2 and 3.8.1.5. One Note permits an engine prelube period prior to diesel generator starting to minimize wear on moving parts which are not lubricated unless the engine is operating. This Note pertains to SR 3.8.1.2 and SR 3.8.1.5. Another Note modifies SR 3.8.1.2 to allow idling and gradual acceleration to minimize mechanical stress and wear on the engine. If these warmup procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.5 must be met when performing SR 3.8.1.2. Another Note modifies SR 3.8.1.2 by allowing performance of SR 3.8.1.5 to satisfy performance of SR 3.8.1.2. Since SR 3.8.1.5 does not allow engine warmup and does require 20-second starting, it is more restrictive than SR 3.8.1.2 and it may be performed in lieu of SR 3.8.1.2. Another Note modifies SR 3.8.1.5 to require performance of SR 3.8.1.3 (diesel generator load test) after performance of SR 3.8.1.5, unless SR 3.8.1.5 is being performed as required by SR 3.8.1.2.

SR 3.8.1.3

This surveillance demonstrates that the diesel generators are capable of synchronizing and accepting $\geq$ the equivalent of the maximum expected accident loads. The 60-minute run time for the diesel generator (required by Ref. 11) is to stabilize the engine temperature. This will ensure that cooling and lubrication are adequate for extended periods of operation while minimizing the time that the diesel generator is connected to the offsite power source.

The normal 31-day Frequency for this surveillance (see Diesel Generator Test Schedule) is consistent with Regulatory Guide 1.108 (Ref. 11).

This surveillance is modified by three Notes. The first Note allows gradual (manual) loading as recommended by the manufacturer to minimize stress and wear on the diesel engine (Ref. 9). The second Note allows momentary transients due to changing bus loads to not invalidate the test. The third Note requires that this surveillance be conducted on only one diesel generator at a time. This will avoid a total loss of AC power due to a common cause failure in the offsite circuits or a perturbation on the grid.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.4

This surveillance verifies that without the aid of the compressor, sufficient air start capacity for diesel generator engine start is available. The system design requirement provides for five engine starts for each air receiver (Ref. 13). However, only one start is assumed for all safety analyses.

Requiring an air pressure ≥ [180] psig provides adequate margin to the diesel generator engine start lockouts. The 31-day Frequency is based on engineering judgment and industry-accepted practice and has been demonstrated adequate for maintaining diesel generator start capability. This Surveillance Requirement is not intended to allow routine operation below the design air capacity represented by [180] psig.

SR 3.8.1.6

Transfer of each 4160 volt emergency bus power supply from the normal preferred offsite circuit to the backup preferred offsite circuit demonstrates the OPERABILITY of the backup circuit distribution network to feed the shutdown loads. The Frequency of the surveillance is based on engineering judgment taking into consideration the plant conditions required to perform the surveillance, and is intended to be consistent with expected fuel cycle lengths.

This surveillance is modified by two Notes. The first Note prohibits performance of this surveillance in MODES 1 or 2. Performance of this surveillance could result in perturbations to the electrical distribution system and cause a challenge to continued steady-state operation in MODES 1 and 2. Therefore, this surveillance must be performed in MODES 3, 4, 5, or 6. The second Note allows credit to be taken for unplanned events in MODES 1 or 2 which satisfy this Surveillance Requirement.

SR 3.8.1.7

The diesel generators are provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed which, if excessive, might result in a trip of the engine. This surveillance demonstrates the diesel generator load response characteristics and capability to reject the largest

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.7 (continued)

a Component Cooling Water Pump

single load without exceeding predetermined voltage and frequency limits, which maintains a specified margin to the overspeed trip. The largest single load on the emergency buses corresponds to ~~Station Service Water Pumps~~ ~~(1250 hp, 935 kW)~~ (1250 BHP, 1037 Kw) (Ref. 4). As required by IEEE-308 (Ref. 14), the load rejection test is acceptable if the increase in the speed of the diesel does not exceed 75% of the difference between nominal speed and the overspeed trip setpoint, or 15% above nominal, whichever is lower. This represents 63 Hz, equivalent to 75% of the difference between nominal speed and the overspeed trip setpoint.

The time, voltage, and frequency tolerances specified in SR 3.8.1.7.b and SR 3.8.1.7.c are derived from Regulatory Guide 1.9 (Ref. 2) recommendations for response during load sequence intervals. The voltage and frequency specified are consistent with the design range of the equipment powered by the diesel generator. SR 3.8.1.7.a corresponds to the maximum frequency excursion while SR 3.8.1.7.b and SR 3.8.1.7.c are steady state voltage and frequency values that the system must recover to following load rejection. The 18-month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 11).

SR 3.8.1.8

This surveillance demonstrates the diesel generator capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The generator full load rejection may occur due to a system fault or inadvertent breaker tripping. This surveillance verifies proper engine-generator load-response under the simulated test conditions. This test will simulate the loss of the total connected loads that the diesel generator will experience following a full load rejection and verify that the diesel generator will not trip upon loss of the load. These acceptance criteria provide for diesel generator damage protection. While the diesel generator is not expected to experience this transient during an event and continue to be available, this response will assure the diesel generator is not degraded for future applications. The 18-month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 11) (expected fuel cycle lengths).

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.8 (continued)

This surveillance is modified by one Note. The Note prohibits performance of this surveillance in MODES 1, 2, 3, 4 or 6. Performance of this surveillance could result in perturbations to the electrical distribution system and cause a challenge to continued steady-state operation in MODES 1 and expose unnecessary risk in Modes 2, 3, 4 and 6. Therefore, this surveillance must be performed in MODE 5, or when no fuel is in the reactor vessel.

SR 3.8.1.9

As required by Regulatory Guide 1.108 (Ref. 11), this surveillance demonstrates the as-designed operation of the standby power sources during loss of the preferred offsite power source. This test verifies all actions encountered from the loss of offsite power including shedding of the non-essential loads and energization of the emergency buses and respective loads from the diesel generator. It further demonstrates the capability of the diesel generator to automatically achieve the required voltage and frequency within the specified time.

The diesel generator automatic start time of 20 seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. The minimum steady state output voltage of 3744 volts is 90% of the nominal 4160 volt output voltage. This value, which is specified in ANSI C84.1-1982, allows for voltage drop down to the terminals of 4000 volt rated motors whose minimum operating voltage is specified as 90% or 3600 volts. It also allows for voltage drops to motors and other equipment down through the 120 volt level where minimum operating voltage is also usually specified as 90% of nameplate rating.

The specified maximum steady state output voltage of 4400 volts is equal to the maximum operating voltage specified for 4000 volt rated motors (+ 10% of motor nameplate rating of 4000 volts). It ensures that for a lightly loaded distribution system the voltage at the terminals of 4000 volt motors will be no more than the maximum rated operating voltages.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.9 (continued)

The specified minimum and maximum steady state output frequency of the diesel generator is 58.8 Hz and 61.2 Hz, respectively. This is equal to ± 2% of the 60 Hz nominal frequency and is derived from the recommendations given in Regulatory Guide 1.9 (Ref. 2) that the frequency should be restored to within 2% of nominal following a load sequence step. The surveillance should be continued for a minimum of five minutes in order to demonstrate all starting transients have decayed and stability has been achieved.

For the purpose of this test, the diesel generators shall be started from standby conditions. Standby conditions in this case means the diesel engine coolant and oil are being continuously circulated and temperature maintained consistent with manufacturer recommendations.

The Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 11), and takes into consideration plant conditions required to perform the surveillance, and is intended to be consistent with expected fuel cycle lengths.

This surveillance is modified by three Notes. The first Note permits an engine prelube period prior to diesel generator starting to minimize wear on moving parts which are not lubricated unless the engine is operating. The second Note prohibits performance of this surveillance in MODES 1, 2, 3, 4 or 6. Performance of this surveillance requires that offsite power be removed from the 4160V emergency buses which will perturb the electrical distribution system and could challenge safety-related equipment. Therefore, this surveillance must be performed in MODE 5 or when no fuel is in the reactor vessel. The third note allows recalibration of individual sequence timers to suffice for retest requirements. This reduces wear on the diesel generators.

SR 3.8.1.10

This surveillance demonstrates that the diesel generator automatically starts and achieves the required voltage and frequency within the specified time (20 seconds) from the design basis activation signal. (It further demonstrates that during a LOOP event, the diesel generator load sequences restart equipment

(continued)

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BASES

SURVEILLANCE REQUUIREMENTS (continued)

SR 3.8.1.10 (continued)

that was deenergized as a result of the LOOP. The five-minute period provides sufficient time to demonstrate stability. The basis for the time, voltage, and frequency tolerances specified in this surveillance are discussed in the Bases for SR 3.8.1.9.

For the purpose of this test, the diesel generators shall be started from standby conditions. Standby conditions in this case means the diesel engine coolant and oil are being continuously circulated and temperature maintained consistent with manufacturer recommendations.

The Frequency of the surveillance is based on engineering judgment taking into consideration the plant conditions required to perform the surveillance and is intended to be consistent with expected fuel cycle lengths.

This surveillance is modified by three ~~M~~Notes. The first Note permits an engine prelube period prior to diesel generator starting to minimize wear on moving parts which are not lubricated unless the engine is operating. The second Note prohibits performance of this surveillance in MODES 1, 2, 3, 4, or 6. Performance of this surveillance requires that certain Safety Injection System (SIS) functions be disabled (e.g., to prevent injection into the reactor vessel) and could challenge continued steady-state operations. Therefore, this surveillance must be performed in MODE 5 or when no fuel is in the reactor vessel. The third note allows recalibration of individual sequence timers to suffice for retest requirements. This reduces wear on the diesel generators.

SR 3.8.1.11

This surveillance demonstrates that the diesel generator automatically starts and achieves the required voltage and frequency within the specified time (20 seconds) from the design basis actuation signal (LOCA signal) and operates for ≥ five minutes. The five-minute period provides sufficient time to demonstrate stability. The basis for the time, voltage, and frequency tolerances specified in this surveillance are discussed in the Bases for SR 3.8.1.9. SR 3.8.1.11.d and SR 3.8.1.11.e ensure that permanently-connected loads and emergency loads are energized from the offsite power system on an ESF signal without loss of offsite power.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.11 (continued)

For the purpose of this test, the diesel generators shall be started from standby conditions. Standby conditions in this case means the diesel engine coolant and oil are being continuously circulated and temperature maintained consistent with manufacturer recommendations.

The Frequency of the surveillance is based on engineering judgment taking into consideration the plant conditions required to perform the surveillance and is intended to be consistent with expected fuel cycle lengths.

This surveillance is modified by three Notes. The first Note permits an engine prelube period prior to diesel generator starting to minimize wear on moving parts which are not lubricated unless the engine is operating. The second Note prohibits performance of this surveillance in MODES 1, 2, 3, 4, or 6. Performance of this surveillance requires that certain Safety Injection (SIS) functions be disabled (e.g., to prevent injection into the reactor vessel) and could challenge continued steady-state operations. Therefore, this surveillance must be performed in MODE 5. The third note allows recalibration of individual sequenct timers to suffice for retest requirements. This reduces wear on the diesel generators.

or when no fuel is in the reactor vessel.

SR 3.8.1.12

In the event of a design basis accident coincident with a loss of offsite power (LOOP), the diesel generators are required to supply the necessary power to ESF Systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

This surveillance demonstrates the diesel generator operation as discussed in the Bases for SR 3.8.1.11 during a LOOP actuation test signal in conjunction with an ESF actuation signal. The Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 11), and takes into consideration plant conditions required to perform the surveillance, and is intended to consistent with expected fuel cycle lengths.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.12 (continued)

This surveillance is modified by three Notes. The first Note permits an engine prelube period prior to diesel generator starting to minimize wear on moving parts which are not lubricated unless the engine is operating. The second Note prohibits performance of this surveillance in MODES 1, 2, 3, 4 or 6. Performance of this surveillance requires that offsite power be removed from the 4160V emergency buses which will perturb the electrical distribution system and could challenge continued steady-state operation and safety-related equipment. Also, certain SIS functions are required to be disabled (e.g., to prevent injection into the reactor vessel). Therefore, this surveillance must be performed in MODE 5 or when no fuel is in the reactor vessel. The third note allows recalibration of individual ~~sequence~~ sequencer timers to suffice for retest requirements. This reduces wear on the diesel generators.

retest methodology

SR 3.8.1.13

This surveillance demonstrates that diesel generator non-critical protective functions (e.g. high jacket water temperature) are bypassed ~~on a loss of voltage signal concurrent with~~ as a result of an ESF actuation test signal. It also verifies that critical protective functions (engine overspeed, generator differential current, Generator Voltage Controlled Overcurrent and low lube oil pressure) trip the diesel generator to avert substantial damage to the diesel generator unit. The non-critical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This provides the operator with sufficient time to react appropriately. The diesel generator availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the diesel generator.

The 18-month Frequency is consistent with other testing performed on the diesel generators during MODES 5 ~~or 6~~ or when no fuel is in the reactor vessel and is based on engineering judgment taking into consideration plant conditions required to perform the surveillance, and is intended to be consistent with expected fuel cycle lengths.

This surveillance is modified by one Note. This Note prohibits performance of this surveillance in MODES 1, 2, 3, 4 or 6. Performance of this surveillance results in diesel generator inoperability and could challenge safety-related equipment. Therefore, this surveillance must be performed in MODE 5 or when no fuel is in the reactor vessel.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.14

Regulatory Guide 1.108 (Ref. 11), requires demonstration once per 18 months that the diesel generators can start and run continuously at full load capability for an interval of not less than 24 hours, of which 22 hours are at a load equivalent to the continuous rating of the diesel and two hours at a load equivalent to the two hour rating of the diesel. The diesel starts for this surveillance can be performed either from cold, standby or hot conditions. The Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 11), and takes into consideration plant conditions required to perform the surveillance and is intended to be consistent with expected fuel cycle lengths.

The load band is provided to avoid routine overloading of the diesel generator. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain diesel generator OPERABILITY.

This surveillance is modified by two Notes. The first Note allows momentary transients due to changing bus loads to not invalidate the test. The second Note prohibits performance of this surveillance in MODES 1, 2, 3, 4 or 6, since it results in a loss of independence between the diesel generator and offsite power for an extended period of time. This increases the risk of a loss of all AC power. Therefore, this surveillance must be performed in MODE 5 or when no fuel is in the reactor vessel

SR 3.8.1.15

This surveillance demonstrates that the diesel engine can restart from a hot condition and achieve the required voltage and frequency within 20 seconds. The 20 second time is derived from the requirements of the accident analysis to respond to a design basis large break LOCA. The bases for the voltage and frequency tolerances are discussed in the Bases for SR 3.8.1.9.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.15 (continued)

This surveillance demonstrates the diesel generator capability to respond to accident signals while hot, such as subsequent to shutdown from normal surveillances. The load band is provided to avoid routine overloading of the diesel generator. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain diesel generator OPERABILITY. The 18-month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 11).

This surveillance is modified by three Notes. The first Note requires that this surveillance be performed within five minutes of shutting down the diesel generator after it has operated for ≥ two hours at fully loaded conditions. The two-hour time limit is based on the manufacturer's recommendation for achieving hot conditions. The second Note permits an engine prelube period prior to diesel generator starting to minimize wear on moving parts which are not lubricated unless the engine is operating. The third Note allows momentary transients due to changing bus loads to not invalidate the test.

SR 3.8.1.16

As required by Regulatory Guide 1.108 (Ref. 11), this surveillance assures that the manual synchronization and manual load transfer from the diesel generator to the offsite power source can be made and the diesel generator can be returned to ready-to-load status when offsite power is restored. It also ensures that the auto-start logic is reset to allow the diesel generator to reload if a subsequent loss of offsite power occurs. The diesel generator is considered to be in "ready-to-load" status when the diesel generator is at rated speed and voltage, the output breaker is open and can receive an auto-close signal on bus undervoltage, and the load sequence timers are reset. The Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 11) and takes into consideration plant conditions required to perform the surveillance.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.16 (continued)

This surveillance is modified by one Note. The Note prohibits performance of this surveillance in MODES 1, 2, 3, 4 or 6. During performance of this test, the offsite circuit has to be removed from service and switching may result in a loss of power to the 4160 volt emergency bus. This would challenge continued steady-state operation and could challenge safety-related systems. Therefore, this surveillance must be performed in MODE 5_X or when no fuel is in the reactor vessel.

SR 3.8.1.17

Demonstration of the test mode override ensures that the diesel generator availability under accident conditions will not be compromised as the result of testing. Interlocks to the LOCA sensing circuits cause the diesel generator to automatically reset to ready-to-load operation if a LOCA actuation signal is received during operation in the test mode. Ready-to-load operation is defined as the diesel generator running at rated speed and voltage with the diesel generator output breaker open. These provisions for automatic switchover are required by IEEE-308 (Ref. 14).

The Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 11) and takes into consideration plant conditions required to perform the surveillance, and is intended to be consistent with expected fuel cycle lengths.

This surveillance is modified by one Note. This Note prohibits performance of this surveillance in MODES 1, 2, 3, 4 or 6. Performance of this test has the potential to perturb the electrical distribution system challenging continued steady-state operation. This test also requires disabling certain SIS functions (e.g., to prevent injection into the reactor vessel). Therefore, this test must be performed in MODE 5_X or when no fuel is in the reactor vessel.

SR 3.8.1.18

As required by Regulatory Guide 1.108 (Ref. 11), each diesel generator is required to demonstrate proper operation for the DBA loading sequence to ensure that voltage and frequency are maintained within the required limits. Under accident conditions, prior to connecting the diesel generators to their appropriate bus, all loads are shed except load center feeders and those motor

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.18 (continued)

control centers which feed Class 1E loads (referred to as permanently-connected loads). Upon reaching 90% rated voltage and frequency, the diesel generators are then connected to their respective bus. Loads are then sequentially connected to the bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers so as to prevent overloading the diesel generators due to high motor starting currents. The 10% load sequence time interval tolerance ensures sufficient time exists for the diesel generator to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 4 provides a summary of the automatic loading of ESF buses. If sequencer timers are found out-of-tolerance, approved calibration checks following timer recalibration will suffice for retest purposes.

The Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 11), and takes into consideration plant conditions required to perform the surveillance and is intended to be consistent with expected fuel cycle lengths.

This surveillance is modified by two Notes. The first Note prohibits performance of this surveillance in MODES 1, 2, 3, or 4. Performance of this test requires the inoperability of certain ESF equipment and has the potential to perturb the electrical distribution system which would challenge continued steady-state operation. Therefore, this test must be performed in MODES 5 ~~or 6~~, The second note allows recalibration of individual sequenct timers to suffice for retest requirements. This reduces wear on the diesel generators. or when no fuel is in the reactor vessel.

SR 3.8.1.19

This surveillance demonstrates that the diesel generator starting dependence has not been compromised. Also, this surveillance demonstrates that each engine can achieve proper speed within the specified time when the diesel generators are started simultaneously. The ten-year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 11) and Regulatory Guide 1.137 (Ref. 12).

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.19 (continued)

This surveillance is modified by a Note which allows an engine prelube period prior to diesel generator starting to minimize wear on moving parts which are not lubricated unless the engine is operating.

Diesel Generator Test Schedule

The diesel generator test schedule (Table 3.8.1-1) implements the recommendations of Revision 3 to Regulatory Guide 1.9 (Ref. 2). The purpose of this test schedule is to provide timely test data to establish a confidence level associated with the goal to maintain diesel generator reliability above 0.95 per demand.

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Per Regulatory Guide 1.9, Revision 3, each diesel generator unit should be tested at least once every 31 days. Whenever a diesel generator has experienced ~~four~~ two or more valid failures in the last ~~25~~ demands, the time between tests is reduced to seven days. ~~Four~~ two failures in ~~25~~ demands is a failure rate of ~~0.16~~ .20, or the threshold of acceptable diesel generator performance, and hence may be an early indication of degradation of the reliability of a diesel generator. However, when considered in the light of a long history of tests, ~~four~~ two failures in the last ~~25~~ demands may only be a statistically probable distribution of random events. Increasing the test frequency will allow for a more timely accumulation of additional test data upon which to base judgment of the reliability of the diesel generator. The increased test frequency must be maintained until seven consecutive failure free tests have been performed.

Regulatory Guide 1.108 (Ref. 11) defines the diesel generator unit as consisting of the engine, generator, combustion air system, cooling water system up to the supply, fuel oil supply system, lubricating oil system, starting energy sources, auto start controls, manual controls, and the diesel generator breaker. Inoperabilities of diesel generators caused by failures of equipment that are not part of the defined diesel generator unit are categorized as invalid failures in accordance with Regulatory Guide 1.108 since the failure would not have prevented the diesel generator from performing its intended safety function. As such, they do not impact the surveillance frequency of the diesel generator that failed.

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BASES

REFERENCES

1. General Design Criteria 17, "Electric Power Systems."

2. Regulatory Guide 1.9, "Selection, Design, and Qualification of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 3 (DRAFT).

3. CESSAR-DC, Section 8.3.1.1.4.

4. CESSAR-DC, Tables 8.3.1-2, and 8.3.1-3.

5. 52 FR 3788, "NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", February 6, 1987.

6. CESSAR-DC, Section 8.3.1.4.

7. CESSAR-DC, Section 8.3.1.2.1.

8. Regulatory Guide 1.93, "Availability of Electric Power Sources," December 1974.

9. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.

10. 10 CFR 50, General Design Criteria 18, "Inspection and Testing of Electric Power Systems."

11. Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as On-site Electric Power Systems at Nuclear Power Plants," August 1977.

12. Regulatory Guide 1.137, "Fuel Oil Systems for Standby Diesel Generators," October 1979.

13. CESSAR-DC, Section 9.5.6.3.

14. IEEE-308 1978, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

16A.11.2 B 3.8.2 AC SOURCES - SHUTDOWN

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

BACKGROUND	A description of offsite and onsite AC Power Sources is provided in the Bases for LCO 3.8.1, AC Sources - Operating.
APPLICABLE SAFETY ANALYSES	In MODES 1, 2, 3, and 4, the requirement to have two independent offsite power sources and two independent onsite standby power sources (diesel generators) OPERABLE is consistent with the initial assumptions of the accident analyses and the design requirements and ~~are~~ is based upon (1) maintaining at least one of the AC and the corresponding DC power sources and associated distribution systems OPERABLE during accident conditions in the event of (1) an assumed loss of all offsite or all onsite AC power, and (2) a single failure of the other AC source. In MODES 5 and 6, the design basis accident (DBA) assumptions of an accident (LOCA) coincident with a loss of offsite or onsite AC power and a single failure are not required to be met. The requirement to have one independent offsite power source and one independent onsite standby power source OPERABLE is adequate to assure power is available in MODES 5 and 6 and when handling irradiated fuel, to systems required to recover from an inadvertent draindown of the reactor vessel or a fuel handling accident (Ref. 2). The OPERABILITY of the minimum specified AC and DC (one offsite power source and one onsite standby power source) power sources and associated distribution systems during shutdown and refueling ensures that (1) the facility can be maintained in the shutdown or refueling condition for extended time periods, and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status. The CESSAR-DC only addresses bounding analyses, such that a specific design basis is not always stated for operation in MODES 5 and 6. The safety analysis assumptions for a design basis event may not be applicable to operation in MODES 5 and 6.

(continued)

BASES

LCO

In MODES 5 and 6 and when handling irradiated fuel, one independent offsite power source between the offsite transmission network and the onsite Class 1E Power Distribution System and one independent onsite standby power source are required to be OPERABLE. This ensures the availability of sufficient power to recover from postulated events in MODES 5 and 6 and when handling irradiated fuel (e.g., fuel handling accident, reactor vessel draindown).

Inoperable AC Sources do not necessarily result in inoperable components (which are designed to receive power from that source) unless specifically directed by Required Actions (refer to LCO 3.0.7).

Certain diesel generator support systems are addressed in other LCOs. During inoperabilities in these support systems, inoperable diesel generators do not necessarily result unless specifically directed by Required Actions (refer to LCO 3.0.7).

APPLICABILITY

The AC power sources that are required to be OPERABLE in MODES 5 and 6 and when handling irradiated fuel provides assurance that:

1. Adequate coolant inventory makeup is available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel,

2. Systems needed to mitigate a fuel handling accident are available, and

3. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown or refueling condition.

AC power requirements for MODES 1, 2, 3, and 4 are addressed in LCO 3.8.1, AC Sources - Operating.

BASES

ACTIONS

A.1, A.2, A.3, A.4, and A.5

With less than the required AC sources OPERABLE, such that only one offsite circuit or one onsite standby power source is OPERABLE, the availability of AC power is degraded and the plant is more susceptible to situations which may lead to complete loss of AC power.

If applicable, CORE ALTERATIONS and handling of irradiated fuel must be immediately suspended. 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 operations with a potential to drain the reactor vessel and subsequent potential for fission product release. Also, if applicable, actions must be initiated immediately to suspend any operations involving positive reactivity additions. Actions must continue until the operations are suspended. These actions preclude the occurrence of the postulated events.

The requirement to immediately initiate action to restore the required AC sources to OPERABLE status is to be continued until restoration of the required AC sources is completed. This is to minimize the time the plant is in this degraded condition of potentially (or actually) not being capable of recovering from the postulated fuel handling accidents or reactor vessel draindown event.

The Completion Times are based on engineering judgment taking into consideration that a time of as short a duration as is practical should be specified while ensuring that the activities are suspended in a controlled manner.

SURVEILLANCE REQUIREMENTS

The Bases provided for SR 3.8.1.1, SR 3.8.1.2, SR 3.8.1.4, SR 3.8.1.5, SR 3.8.1.9, and SR 3.8.1.18 in the Bases for LCO 3.8.1, AC Sources - Operating, are applicable. The performance of the other Surveillance Requirements of LCO 3.8.1, AC Sources - Operating, are not required to be performed in MODES 5 and 6 and when handling irradiated fuel as their performance may adversely affect the reliability of the only required OPERABLE diesel generator.

BASES

REFERENCES

1. NRR Memorandum, Dennis M. Crutchfield to Distribution, Subject: Technical Specifications OPERABILITY Requirements, dated July 8, 1985.

2. CESSAR-DC, Section 15.1.41

3. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

4. NRC Generic Letter No. 88-17, Loss of Decay Heat Removal, October 17, 1988.

16A.11.3 B 3.8.3 DIESEL FUEL AND LUBRICATING OIL

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel and Lubricating Oil

BASES

BACKGROUND

For the purpose of this LCO, the diesel fuel oil subsystem is considered to include 1) fuel oil storage, 2) fuel oil transfer capabilities, and 3) fuel oil properties.

Each diesel generator is provided with a storage tank having a fuel capacity sufficient to operate that diesel for a period of seven days while the diesel generator is supplying maximum post-accident load demand (Ref. 1). The maximum load demand is calculated using the assumption that two diesel generators are available. This onsite fuel capacity is sufficient to operate the diesel generator for longer than the time it would take to replenish the onsite supply from outside sources.

A [1000] gallon day tank is provided for each diesel. Each day tank is housed in a separate room in the diesel generator building and has a fuel capacity for approximately four hours of full load operation (Ref. 1). Fuel oil is transferred from the storage tanks to the day tank by ~~a pump located at each storage tank. Redundancy of pumps and piping precludes the failure of one pump or the rupture of any pipe, valve, or tank to result in the loss of more than one diesel generator.~~ gravity feed

During operation of the diesel generator, fuel oil pumps driven by the diesel engine receive fuel gravity-fed from the day tank and provide fuel to the diesel engine fuel manifolds. Level controls mounted on the day tank automatically ~~start~~ open and ~~stop the associated~~ close storage tank transfer ~~pump~~ valve.

The level of the fuel supply in each storage tank is indicated in the control room. In addition, alarms, both locally and in the control room, annunciate low level and high level in ~~any~~ the associated day tank.

In the unlikely event of a failure in one of the supply trains, the associated day tank low-level alarm annunciates when the fuel oil remaining in the tank provides approximately two hours of full load operation, thus allowing the operator to take corrective action to prevent the loss of the diesel.

(continued)

BASES

BACKGROUND (continued)

For proper operation of the standby diesel generators, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195-1976 (Ref. 3). The fuel oil properties governed by these Surveillance Requirements are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level.

The diesel generator lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated diesel generator under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. The system provides oil to the engine surfaces at a specified temperature during the long anticipated periods of standby duty. Each engine oil sump is of adequate size to contain all the oil in the engine lube oil system and has an inventory capable of supporting a minimum running time of three days. This provides sufficient supply to allow the operator to replenish lube oil from storage facilities onsite. The onsite storage in addition to the engine oil sump is sufficient to ensure seven days continuous operation.

APPLICABLE SAFETY ANALYSES

The initial conditions of design basis transient and accident analyses in CESSAR-DC Chapters 6, Engineered Safety Features, and 15, Accident Analyses, assume ESF systems are OPERABLE. The diesel generators are designed to provide sufficient capacity, capability, redundancy and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for LCO Sections 3.2 (Power Distribution Limits), 3.4 (Reactor Coolant System), and 3.6 (Containment Systems).

The diesel fuel oil subsystem and lubricating oil provide the necessary supply to support operation of the diesel generators.

BASES

LCO

The diesel fuel oil subsystem is required to be OPERABLE and sufficient lubricating oil supply available to ensure availability of the required AC power to shutdown the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated design basis accident.

Diesel fuel oil and lubricating oil not within limits does not necessarily result in inoperable components unless specifically directed by Required Actions (refer to LCO 3.0.7).

APPLICABILITY

The diesel fuel oil subsystem is required to be OPERABLE and sufficient lubricating oil supply is required to be available when the associated diesel generator is required to be OPERABLE (refer to LCO 3.8.1, AC Sources - Operating, and LCO 3.8.2, AC Sources - Shutdown).

ACTIONS

A.1

With the fuel oil level in the day tank low, insufficient fuel oil is available to satisfy the design basis requirement of supporting two hours of continuous operation. However, if the day tank level is between [450] and [900] gallons, sufficient fuel is available to support approximately one hour of continuous diesel generator operation at full load conditions. Consequently, a short time (one hour) is allowed to restore the fuel level. This one-hour Completion Time is acceptable given the low probability of a diesel start requirement during this time period and the automatic start of the storage tank transfer pump on low-level.

B.1

With the fuel oil transfer capability inoperable, sufficient fuel oil is available in the day tank to support two hours of continuous diesel generator operation. Consequently, a short time (one hour) is allowed to restore the transfer capability. This one-hour Completion Time is acceptable given the low probability of a diesel start during this time period. Additionally, this condition is only allowed to exist on one diesel generator at a time. Therefore, one diesel generator will still retain full fuel oil transfer capability.

(continued)

ACTIONS (continued)

C.1

Low level in one or more fuel oil storage tanks indicates that the design basis requirement of supporting seven days of continuous operation may not be able to be satisfied. However, if the storage tank level is between [25,000] and [55,000] gallons, sufficient fuel is available to support approximately three days of continuous diesel generator operation at full load conditions. Consequently, a period of 24 hours is allowed to restore the fuel level. This 24-hour Completion Time is acceptable given the likelihood that additional fuel could be provided from offsite or transferred between storage tanks as necessary in the unlikely occurrence of a loss of offsite power event during this time period.

D.1

With lubricating inventory less than [ ] gallons, sufficient lubricating oil to support seven days of continuous diesel generator operation at full load conditions may not be available. However, a limited time (24 hours) is provided to restore the lubricating oil inventory because of the relatively low rate of usage and the likelihood that additional supplies can be provided from offsite.

E.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.5 not within the required limits, a period of 14 days is allowed to restore the stored fuel oil properties. This provides sufficient time to retest the fuel oil sample to confirm that the initial test results were valid, test the stored fuel oil to determine that the new fuel, when mixed with previously stored fuel, remains acceptable, and/or restore the stored fuel properties. This restoration may involve feed-and-bleed procedures, filtering, or combinations of these and other procedures. Even if a diesel generator start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the diesel generator would still be capable of performing its intended function.

(continued)

BASES

ACTIONS (continued)

F.1

For the particulate contamination testing portion of the diesel fuel oil testing program, a period of 30 days is provided to restore the fuel oil properties to within limits. This test demonstrates the total particulate in stored fuel are less than 10 mg/liter. Fuel oil with total particulate concentrations exceeding the 10 mg/liter will impact diesel generator performance on a long term basis; therefore, the 30-day Completion Time will not adversely affect diesel generator reliability and is acceptable.

G.1

With the Required Actions and associated Completion Times not met, or the diesel fuel oil subsystem inoperable for reasons other than addressed by Conditions A through F, the associated diesel generator may be incapable of performing its intended function and must be immediately declared inoperable. The Required Actions of LCO 3.8.1, AC Sources - Operating, or LCO 3.8.2, AC Sources - Shutdown, as applicable, must then be followed.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

The verification of each diesel engine fuel day tank supply ensures that enough fuel is on hand at the engine to sustain at least one hour of full load operation for the diesel generator (Ref. 1). The 31-day Frequency is adequate to ensure a sufficient fuel supply is available since operators should be aware of large uses during this period and low-level alarms are provided. As such, the Frequency is based on engineering judgment.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.2

The verification of each diesel engine fuel storage tank supply demonstrates that sufficient fuel is available to sustain at least seven days of full load operation for the diesel generator (Ref. 1). This time period is sufficient to place the unit in a safe shutdown condition and provides enough time to bring in replenishment fuel from an offsite location. The 31-day Frequency is based on engineering judgment and is sufficient to ensure diesel fuel oil availability since operators should be aware of large uses during this period.

SR 3.8.3.3

This surveillance ensures that sufficient lubricating oil inventory is available to support at least seven days of full load operation for the diesel generator. The [ ] gallons requirement is based on the diesel generator manufacturer's consumption values for the run-time of the diesel. A 31-day Frequency is adequate to ensure a sufficient lubricating oil supply is onsite since diesel generator starts and run times are closely monitored by the plant staff.

SR 3.8.3.4

The tests listed below are a means of determining whether or not new fuel has been contaminated with substances which would have an immediate, detrimental impact on diesel engine combustion/operation. If results from these tests are within acceptable limits, the fuel may be added to the storage tanks without concern for contaminating the entire volume of fuel in the storage tanks. The tests, limits, and applicable ASTM standards are as follows:

a. Sample new fuel in accordance with ASTM D270-(81).

b. Verify in accordance with tests specified in ASTM D975-(82) that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 but ≤ 0.89 or an API gravity at 60°F of ≥ 27° but ≤ 39°, a kinematic viscosity at 40°C of ≥ 1.9 centistokes but ≤ 4.1 centistokes, and a flash point ≥ 125°F.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.4 (continued)

c. Verify the new fuel has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-(86).

These tests are required to be performed within 31 days prior to adding fuel to the storage tanks. The Frequency is established by Regulatory Guide 1.137 (Ref. 2). Failure to meet any of the above limits is cause to reject the new fuel, but does not constitute a diesel generator OPERABILITY concern since the fuel is not added to the storage tanks.

SR 3.8.3.5

Within 31 days following the initial new fuel oil sample, this surveillance is performed to establish that the other properties specified in Table 1 of ASTM D975-(82) are met for new fuel when tested in accordance with ASTM D975-(82), except that the analysis for sulfur may be performed in accordance with ASTM D1522-(83) or ASTM D2622-(82). The required surveillance Frequency for sampling the fuel oil is based on engineering judgment taking into consideration the likelihood of a change in fuel oil parameters.

SR 3.8.3.6

This surveillance is an integral part of a comprehensive program to ensure the availability of high quality fuel oil for the diesel generators at all times. By testing for particulate on a 31-day basis, information regarding the condition of stored fuel can be obtained and trended. Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. However, the particulate can cause fouling of filters and fuel injection equipment which can cause engine failure. If particulate is removed from stored fuel by circulating it through filters (other than diesel engine filters), the fuel can be restored to acceptable condition, and its storage life extended indefinitely. By obtaining and trending particulate data, it is possible to determine when stored fuel cleanup will be necessary. This can be done well before the maximum allowable particulate concentration is reached.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.6 (continued)

Particulate concentrations should be determined in accordance with ASTM D2274-(74), Method A. This method involves a gravimetric determination of total particulate concentration in the fuel and has a limit of 10mg/liter. It is acceptable to obtain a field sample for subsequent lab testing in lieu of field testing. In the case(s) where the total stored fuel volume is contained in two or more interconnected tanks, each tank must be considered and tested separately. The Frequency of the surveillance is based on engineering judgment taking into consideration the ease of performing the test and the likelihood of a change in particulate concentrations.

SR 3.8.3.7 and SR 3.8.3.8

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria which can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from fuel day tanks and storage tanks once per 31 days will eliminate the necessary environment for survival. This is the most effective means of controlling microbiological fouling. In addition, it will eliminate the potential for water entrainment in the fuel oil during diesel generator operation.

Water may come from any of several sources including condensation, ground water, rain water, contaminated fuel, and from breakdown of the fuel by bacteria. Frequent checking for and removal of accumulated water will minimize fouling as well as providing data regarding fuel system water tight integrity. The surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 2).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.9

This surveillance demonstrates that each required fuel oil transfer ~~pump~~ valve operates and transfers fuel from its associated storage tank to its associated day tank. This is required to support seven days continuous operation of the standby power sources. This surveillance provides assurance that the fuel transfer pump is OPERABLE, there is no crud buildup blocking the fuel supply line, and that the fuel supply line is intact. The 92-day Frequency corresponds to the Inservice Testing requirements of pumps per ASME Section XI.

SR 3.8.3.10

The draining of the fuel oil in the storage tanks, removal of accumulated sediment, and tank cleaning is required at ten-year intervals by Regulatory Guide 1.137 (Ref. 2). This is required to perform the ASME Section XI examinations of the tanks. To preclude the introduction of surfactants in the fuel system, the cleaning should be accomplished using sodium hypochlorite solutions or their equivalent rather than soap or detergents.

REFERENCES

1. CESSAR-DC, Section 9.5.4.2.

2. Regulatory Guide 1.137, "Fuel Oil Systems for Standby Diesel Generators," October 1979.

3. ANSI N195-1976, "Fuel Oil Systems for Standby Diesel Generators," Appendix B.

4. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

16A.11.4 B 3.8.4 DC SOURCES - OPERATING

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources - Operating

BASES

BACKGROUND

The Class 1E DC Power System provides control power for the AC emergency power system. It also provides both motive and control power to selected safety-related equipment and provides circuit breaker control power for the 600 volts and lower AC distribution system. The DC Power System is also the source of power for the vital instrumentation buses via inverters. The ~~four~~ six DC subsystems conform to the independence and redundancy requirements of Regulatory Guide 1.6 (Ref. 1), IEEE-308 (Ref. 2), and General Design Criteria 17 (Ref. 3). The six batteries are:

DIVISION I	DIVISION II
Division I Battery	Division II Battery
Channel A Battery	Channel B Battery
Channel C Battery	Channel D Battery

Each DC subsystem is energized by a dedicated 125 volt battery and associated 125 volt battery charger. Each battery is exclusively associated with a single 125 volt DC bus~~.~~ and ~~E~~each battery charger is supplied by its associated AC load group only. ~~The battery and the battery charger exclusively associated with a 125 volt DC subsystem cannot be interconnected with any other 125 volt DC subsystem. The charger is supplied from the same AC load group for which the associated DC subsystem supplies the control power. The loads between the redundant 125 volt DC subsystem are not transferable.~~

Each of the ~~four~~ Six DC subsystems is made up of the following:

- A [120-cell lead-calcium battery] rated at [1650] Ah for eight hours to [210] volts at 77°F,

- A static battery charger rated at [400] amps with 0.5% voltage regulation with an AC supply variation of 10% in voltage and 5% in frequency, and

- associated switchboards and distribution panels.

(continued)

However, in order to fulfill the battery capacity criteria: "to supply one division battery's loads and one channel of loads," the batteries MAY be cross-tied to allow coping strategies to be implemented in accordance with the capacity sizing. Additionaly, the batteries provide a SBO coping capability which, assuming manual load shedding or the use of load management programs, exceeds two hours, and as a minimum, permits operating the instrumentation and control loads associated with the turbine-driven emergency feedwater pumps for 8 hours.

BASES

BACKGROUND (continued)

the division battery and one channel of loads for two hours

Battery operating voltage is 125 volts and each battery has adequate storage supply ~~the required loads for four hours~~ without recharging (Ref. 4). Capacity is adequate for all loss of coolant accident (LOCA) conditions or any other emergency shutdown.

Each 125 volt DC Class 1E battery is separately housed in a ventilated room apart from its charger and distribution center. Each subsystem is located in an area separated physically and electrically from other subsystems to ensure that a single failure in one subsystem does not cause failure in the redundant subsystem. In normal alignment, there is no sharing between redundant Class 1E subsystems such as batteries, battery chargers, or distribution panels. Class 1E batteries of the same division may be cross tied together for accident coping (SBO) and/or LCO purposes.

All batteries are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end-of-life-cycles and the 100% design demand. Battery size is based on 100% of required capacity, and after selection of an available commercial battery, results in a battery capacity in excess of 125% of required capacity. The voltage design limit is [ ] volts per cell which corresponds to a total minimum voltage output of [ ] volts per battery bank (Ref. 5).

Each battery charger has ample power-output capacity for the steady-state operation of connected loads required during normal operation while at the same time maintaining its battery bank fully charged. Each battery charger has sufficient capacity to restore the battery bank from the design minimum charge to 95% of its fully charged state in [12] hours while supplying normal steady-state loads (Ref. 4).

BASES

APPLICABLE SAFETY ANALYSES

The initial conditions of design basis transient and accident analyses in CESSAR-DC, Chapters 6, Engineering Safety Features, and 15, Accident Analyses, assume Engineered Safety Features (ESF) systems are OPERABLE. The DC Power System provides normal and emergency DC power for the diesel generators, emergency auxiliaries, and for control and switching during all MODES of operation. The OPERABILITY of the power sources is consistent with the initial assumptions of the accident analyses which are based upon maintaining the required DC power sources and associated distribution systems OPERABLE during accident conditions in the event of (1) an assumed loss of offsite power, and (2) an additional single failure.

LCO

Division 1 and 2 DC Power Sources are required to be OPERABLE to ensure availability of the required power to shutdown the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated design basis accident. Loss of any one of the DC power subsystems does not prevent the minimum safety function from being performed. Each DC Power Source ~~Division~~ is considered OPERABLE if the 125 volt battery and associated battery charger satisfy the applicable Surveillance Requirements.

Inoperable DC sources do not necessarily result in inoperable components unless specifically directed by Required Actions (refer to LCO 3.0.7). The electrolyte parameter limits relationship to the OPERABILITY of DC sources is covered by LCO 3.8.6, Battery Cell Parameters. During periods when battery cell parameters are not within limits, DC sources are not necessarily inoperable unless specifically directed by the Required Actions of LCO 3.8.6, Battery Cell Parameters. This is in accordance with LCO 3.0.7.

BASES

APPLICABILITY

The DC Power Sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to provide power for instrumentation and controls to ensure safe plant operation and to ensure that:

1. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of anticipated operational occurrences or abnormal transients, and

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

DC power requirements for MODES 5 and 6 are addressed in the Bases for LCO 3.8.5, DC Sources - Shutdown.

ACTIONS

A.1 and A.2

(see next page)

~~A~~.1
B

two
With ~~one~~ of the required DC Power Sources ~~[Divisions]~~ inoperable (e.g., inoperable ~~battery~~ batteries, or inoperable battery chargers, or inoperable battery chargers and an associated inoperable ~~battery~~ batteries), the remaining DC Power Sources ~~has~~ have the capacity to support a safe shutdown and to mitigate an accident condition. However, since a subsequent worst case single failure would result in the loss of the 125 volt Class 1E battery system, continued power operation should not exceed two hours. The two-hour Completion Time is based on Regulatory Guide 1.93 (Ref. 7), and engineering judgment considering the number of available systems and the time required to reasonably complete the Required Actions.

C C
~~B~~.1 and ~~B~~.2

The plant must be placed in a MODE in which the LCO does not apply if the DC Power Source cannot be restored to OPERABLE status in the associated Completion Time. This is done by placing the plant in at least MODE 3 in six hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power without challenging plant systems.

Add to page B3.8-43/16A.11-43

A.1 and A.2

With one of the six batteries inoperable (Division I Battery, Channel A Battery, Channel C Battery, Division II Battery, Channel B Battery, Channel D Battery) or one battery charger inoperable, or a combination of battery and associated battery charger inoperable, the battery crossties may be utilized to allow the remaining two operable divisional batteries to power the loads of the inoperable power source and fulfill the SBO coping capability. This is possible since each battery is sized to provide the one division battery's loads and one channel of loads. Thus, the two remaining operable batteries may power the defunct ~~batteries~~ battery's loads while it is being restored to operability. This design feature should be utilized with the intent of restoring the inoperable components as soon as practicable — within 24 hours.

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the 125/250 volt Class 1E battery helps ensure the effectiveness of the charging system and the ability of the battery to perform its intended function. Float charge is the condition where the charger is supplying the continuous charge required to overcome the internal losses of a battery (or batte...) and maintain the battery (or battery cell) in a fully charged state. The seven-day Frequency is consistent with the manufacturers' recommendations and IEEE 450 (Ref. 8). The Frequency is based on engineering judgment and industry-accepted practice considering the unit conditions required to perform the test, the ease of performing the test, and the likelihood of a change in system or component status.

SR 3.8.4.2

Visual inspection of the battery cells and connections or measurement of the resistance of each cell and terminal connection provide an indication of physical damage or abnormal deterioration which could potentially degrade battery performance. The connection resistance value is a ceiling value established by the battery manufacturer based on calculations taking into consideration the physical configuration of the batteries. The 92-day Frequency is sufficient for detecting trends in these conditions indicative of any problems. A more complete inspection is performed in conjunction with the preventive maintenance program conducted during refueling outages.

SR 3.8.4.3

Visual inspection of the battery cells, cell plates, and battery racks provide an indication of physical damage or abnormal deterioration which could potentially degrade battery performance. The 18-month Frequency is based on engineering judgment and operational experience and is sufficient to detect battery degradation on a long-term basis when it is properly coupled with other surveillances more frequently performed to detect abnormalities.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.4 and 3.8.4.5

Visual inspections and resistance measurements of the cell-to-cell and terminal connections provide an indication of physical damage or abnormal deterioration which could indicate degraded battery performance. The anti-corrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal and inspection under each terminal connection. The connection resistance value is a ceiling value established by the manufacturer based on calculations taking into consideration the physical configuration of the batteries. The 18-month Frequency is based on engineering judgment and industry accepted practice taking into consideration the likelihood of a change in system or component status.

SR 3.8.4.6

Regulatory Guide 1.32 (Ref. 9), requires that the battery charger supply be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to 95% of the fully charged state, irrespective of the status of the unit during which these demands occur. The minimum required amperes and duration ensures that the DC load requirements can be satisfied (refer to SR 3.8.4.7). The Frequency is based on engineering judgment and industry accepted practice considering the unit conditions required to perform the test, and is intended to be consistent with expected fuel cycle lengths.

This surveillance is modified by two Notes. The first Note prohibits performance of this surveillance in MODES 1, 2, 3, or 4. Performance of this test requires the associated DC Division to be inoperable during the test. Therefore, this test must be performed in MODES 5 or 6. The second Note allows credit to be taken for unplanned events in MODES 1, 2, 3, or 4 which satisfy this Surveillance Requirement.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.7

Regulatory Guide 1.32 (Ref. 9), requires the performance of a battery service test in accordance with IEEE-450 (Ref. 8) at intervals not to exceed 18 months. A battery service test is a special capacity test to demonstrate the capability of the battery to meet the system analyzed response requirements. Reference 10 provides the load requirements for the batteries.

This surveillance is modified by three notes. The first Note allows the once per 60-month performance of SR 3.8.4.8 in lieu of SR 3.8.4.7. This is allowed since SR 3.8.4.8 represents a more severe test of battery capacity than SR 3.8.4.7. The second Note prohibits performance of this surveillance in MODES 1, 2, 3, or 4. Performance of this test requires the associated DC Division to be inoperable during the test. Therefore, this test must be performed in MODES 5 or 6. The third Note allows credit to be taken for unplanned events in MODES 1, 2, 3, or 4 which satisfy this Surveillance Requirement.

SR 3.8.4.8

IEEE-450 (Ref. 8) recommends a performance test for each battery at 60-month intervals. A battery performance test is a capacity test of the battery in the "as found" condition, after being in service, to detect any change in the capacity as determined by the new battery acceptance test. IEEE-485 (Ref. 11) recommends that the battery should be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows the battery rate of deterioration is increasing even if there is ample capacity to meet the load requirements. The acceptance criteria for this surveillance specifies a more restrictive 85% capacity based on the extension of the Frequency for SR 3.8.4.7 from the IEEE-450 recommendation of 12 months to 18 months.

IEEE-450 (Ref. 8) recommends that, in addition to the 60-month Frequency, a performance discharge test should be performed every 18 months for any battery that shows signs of degradation or has reached 85% of the service life expected of the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.8 (continued)

The Frequencies are consistent with the guidance in Regulatory Guide 1.129 (Ref. 12) and are sufficient to identify trends in battery degradation.

This surveillance is modified by two Notes. The first Note prohibits performance of this surveillance in MODES 1, 2, 3, or 4. Performance of this test requires the associated DC Division to be inoperable during the test. Therefore, this test must be performed in MODES 5 or 6. The second Note allows credit to be taken for unplanned events in MODES 1, 2, 3, or 4 which satisfy this Surveillance Requirement.

REFERENCES

1. Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," March 10, 1971.

2. IEEE-308 1974, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

3. 10 CFR 50, General Design Criteria 17, "Electric Power Systems."

4. CESSAR-DC, Section 8.3.2.1.1.

5. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

6. CESSAR-DC, Section 8.3.2.2.1.

7. Regulatory Guide 1.93, "Availability of Electric Power Sources," December 1974.

8. IEEE-450 1980, "IEEE Recommended Practice for Maintenance Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Subsystems."

(continued)

BASES

REFERENCES (continued)

9. Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," February 1977.

10. CESSAR-DC, Table 8.3.2-4.

11. IEEE-485 1983, Recommended Practices for Sizing Large Lead Storage Batteries for Generating Stations and Substations." June, 1983.

16A.11.5 B 3.8.5 DC SOURCES - SHUTDOWN

B 3.8 ELECTRICAL POWER SYSTEMS

B.3.8.5 DC Sources - Shutdown

BASES

BACKGROUND	A description of the DC Power Sources is provided in the Bases for LCO 3.8.4, DC Sources - Operating.
APPLICABLE SAFETY ANALYSES	In MODES 1, 2, 3, and 4, the requirement to have two independent DC Power Source Divisions OPERABLE is consistent with the initial assumptions of the accident analyses and the design requirements and are based upon: maintaining at least one of the AC and the corresponding DC Power Sources and associated distribution systems OPERABLE during accident conditions in the event of (1) an assumed loss of all offsite or all onsite AC power, and (2) a single failure of the other AC or DC source. In MODES 5 and 6, the design basis accident (DBA) assumptions of a loss of coolant accident (LOCA) coincident with a loss of offsite or onsite AC power and a single failure are not required to be met. Therefore, the requirement to have one DC Power Source Division OPERABLE is adequate to assure power is available in MODES 5 and 6 and when handling irradiated fuel to systems required to recover from an inadvertent draindown of the reactor vessel or a fuel handling accident (Ref. 1). The OPERABILITY of the minimum specified AC and DC Power Sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods, and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status (Ref. 2). The CESSAR-DC only addresses bounding analyses, such that a specific design basis is not always stated for operation in MODES 5 and 6.

(continued)

BASES

LCO	In MODES 5 and 6 and when handling irradiated fuel, one DC Power Source Division is required to be OPERABLE. This ensures the availability of sufficient power to recover from postulated events in MODES 5 and 6 and when handling irradiated fuel (e.g., fuel handling accident, reactor vessel draindown). A description of OPERABILITY requirements for the DC Power Source Division is provided in the Bases of LCO 3.8.4, DC Sources - Operating. The electrolyte parameter limits relationship to the OPERABILITY of DC sources is dictated by LCO 3.8.6, Battery Cell Parameters. During periods when battery cell parameters are not within limits, inoperable DC sources do not necessarily result unless specifically directed by the Required Actions of LCO 3.8.6, Battery Cell Parameters. This is in accordance with LCO 3.0.7.
APPLICABILITY	The DC Power Sources required to be OPERABLE in MODES 5 and 6 and when handling irradiated fuel assures sufficient power to ensure that: 1. Adequate coolant inventory makeup is available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel, 2. Systems needed to mitigate a fuel handling accident are available, and 3. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown or refueling condition. DC power requirements for MODES 1, 2, 3, and 4 are addressed in LCO 3.8.4, DC Sources - Operating.

Amendment I
December 21, 1990

BASES

ACTIONS

A.1, A.2, A.3, A.4, and A.5

With less than the required DC Power Sources OPERABLE, no DC power is available to recover from postulated events in MODES 5 and 6 and when handling irradiated fuel.

If applicable, CORE ALTERATIONS and handling of irradiated fuel must be immediately suspended. 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 operations with a potential to drain the reactor vessel to minimize the probability of a vessel draindown and subsequent potential for fission product release. Also, if applicable, actions must be initiated immediately to suspend any operations involving positive reactivity additions. Actions must continue until the operations are suspended. These actions preclude the occurrence of the postulated events.

The requirement to immediately initiate action to restore the required DC sources to OPERABLE status is to be continued until restoration of the required DC Power Sources is completed. This is to minimize the time the plant is in this degraded condition of potentially (or actually) not being capable of recovering from the postulated fuel handling accidents or reactor vessel draindown event.

The Completion Times are based on engineering judgment taking into consideration that a time of as short a duration as is practical should be specified while ensuring that the activities are suspended in a controlled manner.

SURVEILLANCE REQUIREMENTS

The Bases provided for SR 3.8.4.1 through SR 3.8.4.8 in the in the Bases for LCO 3.8.4, DC Sources - Operating, are applicable.

BASES

REFERENCES

1. Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors, NUREG-0212, Revision 3, dated December 1981.

2. NRR Memorandum, Dennis M. Crutchfield to Distribution, Subject: Technical Specifications OPERABILITY Requirements, dated July 8, 1985.

3. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

16A.11.6 B 3.8.6 BATTERY CELL PARAMETERS

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND

LCO 3.8.6, Battery Cell Parameters, utilizes Table 3.8.6-1 to delineate the limits on electrolyte level, float voltage, and specific gravity for the DC Power Source batteries. A discussion of these batteries and their OPERABILITY requirements are provided in the Bases for LCO 3.8.4, DC Sources - Operating, and LCO 3.8.5, DC Sources - Shutdown. Within this table, Category A defines the limits for each designated pilot cell and Category B does the same for each connected cell.

The Category A limits for the designated pilot cell's float voltage ≥ 2.13 volts and a specific gravity of ≥ 1.200 (0.015 below the manufacturer's fully charged nominal specific gravity) or a battery charging current that had stabilized at a low value) is characteristic of a charged cell with adequate capacity. The limits on electrolyte level ensures no physical damage to the plates occurs and adequate electron transfer capability is maintained in the event of transient conditions.

The Category B limits for each connected cell's float voltage and specific gravity ≥ 2.13 volts and a specific gravity of ≥ 1.195 (0.020 below the manufacturer's fully charged nominal specific gravity with an average specific gravity of all the connected cells ≥ 1.205 (0.010 below the manufacturer's fully charged nominal specific gravity) ensures the OPERABILITY and capability of the battery. The limits on electrolyte level ensure no physical damage to the plates occurs and adequate electron transfer capability is maintained in the event of transient conditions.

The limits are based upon manufacturer's recommended values (Ref. 1) to ensure the OPERABILITY and capability of the battery. The specific gravity limits assure a manufacturer's recommended fully charged nominal specific gravity of 1.215. Specific gravity must be corrected for electrolyte temperature and level, and the float voltage limits may be corrected for average electrolyte temperature. These Notes provide for correction of the measured values in accordance with manufacturer's recommendations when the values reflect transient conditions as opposed to battery capacity.

(continued)

BASES

BACKGROUND (continued)	Category C defines allowable values of electrolyte level, float voltage, and specific gravity of each connected cell. These values represent degraded battery conditions. However, operation is permitted when Category C limits are met since sufficient capacity exists to perform the intended function. These values are discussed in more detail in the Actions section of this Bases.
APPLICABLE SAFETY ANALYSES	The initial conditions of design basis transient and accident analyses in CESSAR-DC Chapters 6, Engineering Safety Features, and 15, Accident Analyses, assume all Engineered Safety Features (ESF) systems are OPERABLE. The DC power systems provide normal and emergency DC power for emergency auxiliaries and for control and switching during all MODES of operation. The OPERABILITY of the power sources is consistent with the initial assumptions of the accident analyses and is based upon maintaining DC power sources and associated distribution systems OPERABLE during accident conditions in the event of (1) an assumed loss of offsite power, and (2) an additional single failure.
LCO	The Class 1E DC Power Sources are required to be OPERABLE to ensure availability of the required power to shutdown the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated design basis accident. Battery cell parameters not within limits do not necessarily result in inoperable DC sources (batteries) unless specifically directed by Required Actions (refer to LCO 3.0.7).
APPLICABILITY	The battery electrolyte must be within the limits of Table 3.8.6-1 when the associated DC Power Sources are required to be OPERABLE (refer to Applicability discussion in Bases for LCO 3.8.4, DC Sources - Operating, and LCO 3.8.5, DC Sources - Shutdown).

BASES

ACTIONS

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

Operation with one or more cells in one or more batteries parameters not within limits (i.e., Category A limits not met, or Category B limits not met, or Category A and B limits not met), but within the allowable value (Category C limits are met) specified in Table 3.8.6-1 is permitted since sufficient capacity exists to perform the intended function. The pilot cell electrolyte level and float voltage are required to be verified to meet the Category C allowable values within one hour (Required Action A.1). This check will provide a quick indication of the status of the remainder of the battery cells. One hour provides time to inspect the electrolyte level and to confirm the float voltage of the [ ] pilot cells. As such, the Completion Time is based on engineering judgment taking into consideration the time required to perform the Required Action.

Verification that the Category C allowable values are met for all cells (Required Action A.2) will ensure that during the time to restore the parameters to the Category A and B limits that the battery will still be capable of performing its intended function. Twenty four hours are provided to complete Required Action A.2 because specific gravity measurements must be obtained for each connected cell. As such, the Completion Time is based on engineering judgment taking into consideration the time required to perform the Required Action and the assurance provided by Required Action A.1 that the battery cell parameters are not severely degraded.

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. During this 31-day period:

(1) the allowable values for electrolyte level (above the top of the plates and not overflowing), ensures no physical damage to the plates with an adequate electron transfer capability;

(continued)

BASES

ACTIONS (continued)

A.1, A.2, and A.3 (continued)

(2) the allowable value for the average specific gravity of all the cells ≥ [1.195 [0.020] below the manufacturer's recommended fully charged nominal specific gravity], or a battery charging current that had stabilized at less than (2) amperes on a float charge) is the manufacturer's recommendation and ensures that the decrease in capacity will be less than the margin provided in sizing;

(3) the allowable value for an individual cell's specific gravity [0.020] below the average of all the connected cells) ensures that an individual cell's specific gravity will not be [0.040] below the manufacturer's fully charged nominal specific gravity. This is the value recommended by the manufacturer to ensure the overall capability of the battery will be maintained within an acceptable limit; and

(4) the allowable value for an individual cell's float voltage [> 2.07] volts) ensures the battery's capability to perform its design function.

The 31-day Completion Time is based on engineering judgment taking into consideration that while battery capacity is degraded, sufficient capacity exists to perform the intended function and allow time to fully restore the battery cell parameters to normal limits.

When any battery parameter is outside the Category C allowable value, sufficient capacity to supply the maximum expected load requirements is not assured and Condition B would be entered.

(continued)

BASES

ACTIONS (continued)

B.1

If the appropriate parameters in Table 3.8.6-1 cannot be met in accordance with the Required Actions and associated Completion Times of Condition A, or Category C allowable values are not met, the associated battery must be declared inoperable immediately and the Required Actions of LCO 3.8.4, DC Sources - Operating, or LCO 3.8.5, DC Sources- Shutdown, followed as appropriate. The battery must also be declared inoperable if the average electrolyte temperature of the pilot cells is less than the limits of SR 3.8.6.3. Below this temperature, the battery's capability may not be sufficient to satisfy the design basis load profile.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This surveillance is consistent with the recommendations of IEEE-450 (Ref. 3), which states that the battery be demonstrated to meet Category A limits on a regularly scheduled interval. The seven-day Frequency for this surveillance is consistent with the guidance of IEEE-308 (Ref. 4).

SR 3.8.6.2

This surveillance is consistent with the recommendations of IEEE-450 (Ref. 3), which states that the battery be demonstrated to meet Category B limits on a quarterly basis. In addition, within 24 hours of a battery discharge below 110 volts or a battery overcharge above 150 volts, the battery must be demonstrated to meet Category B limits. This one-time 24-hour surveillance verifies that no significant degradation of the battery occurred as a consequence of the discharge or overcharge. The 24 hour requirement is judged to be acceptable given the seven day periodic check of the Category A limits.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.6.3

This surveillance is consistent with the recommendations of IEEE-450 (Ref. 3) which states that the battery average electrolyte temperature be determined on a quarterly basis. While higher than normal operating temperatures increase battery capacity, elevated temperatures also increase internal self-discharge, lower cell voltages for a given charge voltage, and shorten overall battery life. Lower than normal temperatures have the opposite effects and act to inhibit or reduce battery capacity. Normal battery operating temperatures are 60°F to 90°F with a recommended ambient operating temperature of 77°F. This surveillance ensures that the ambient temperatures remain within an acceptable operating range. These limits were established based on manufacturer's recommendations.

REFERENCES

1. (Battery Manufacturer recommended parameter values).

2. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

3. IEEE-450 1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

4. IEEE-308 1978, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

16A.11.7 B 3.8.7 DISTRIBUTION SYSTEMS - OPERATING

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Distribution Systems - Operating

BASES

BACKGROUND

The primary distribution of the onsite AC Power Distribution System is at 4160 volts. There are two 4160 volt emergency buses. These buses are located in separate rooms in the diesel building and supply power to essential loads required during planned operations and during abnormal operational transients and accidents. Power is distributed to the 4160 volt buses from the offsite power sources as described in the Bases for LCO 3.8.1, AC Sources - Operating. Control power for the 4160 volt breakers is supplied from the Class 1E batteries as described in the Bases for LCO 3.8.4, DC Sources - Operating.

The secondary plant distribution is at 480 volts. The 480 volt distribution system includes load centers [ ]. Load centers [ ] are normally supplied from 4160 volt buses [ ], respectively, through their own transformers. The 480 volt load centers are located in separate rooms in the control building. Control power for the 480 volt breakers is supplied from the Class 1E batteries as described in the Bases for LCO 3.8.4, DC Sources - Operating.

The safety-related 480 volt AC motor control centers are fed from load centers [ ]. The 120 volt AC vital buses are arranged in four load groups and are normally powered from their 125 volt DC switchboards, respectively via the associated DC/AC inverter. The alternate power supply for the vital buses is a Class 1E constant voltage source powered from the same Division as the associated inverter. The Division 1 constant voltage source is powered from 480 volt motor control center (MCC) [ ] and supplies load group [ ] vital buses. Similarly, MCC [ ] (Division 2) supplies load group [ ] vital buses.

The 125 volt DC load groups 1 through 4 switchboards, respectively normally are powered from their battery charger. The battery chargers are powered from their Divisional 480 volt MCC. A loss of AC power or

(continued)

BASES

BACKGROUND (continued)

failure of the battery charger places the associated battery in service to supply its 125 volt DC switchboard.

The list of all required distribution buses is located in Table B 3.8.7-1.

APPLICABLE SAFETY ANALYSES

The initial conditions of design basis transient and accident analyses in CESSAR-DC Chapters 6, Engineering Safety Features, and 15, Accident Analyses, assume Engineered Safety Features (ESF) systems are OPERABLE. The AC and DC electrical Power Distribution Systems are designed to provide sufficient capacity, capability, redundancy and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for LCO sections 3.2 (Power Distribution Limits), 3.4 (Reactor Coolant System), and 3.6 (Containment Systems).

The OPERABILITY of the Power Distribution Systems is consistent with the initial assumptions of the accident analyses and are based upon maintaining at least one of the onsite AC and DC power sources and associated distribution systems OPERABLE during accident conditions in the event of (1) an assumed loss of all offsite power or all onsite AC power, and (2) a single failure of the other AC or DC source.

LCO

The Power Distribution System Divisions listed in Table B 3.8.7-1 ensure the availability of AC and DC power for the systems required to shutdown the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated design basis accident. Two Divisions of the AC and DC Power Distribution Systems are required to be OPERABLE.

Maintaining two Divisions of AC and DC Power Distribution Systems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Either Division of the distribution system is capable of providing the necessary electrical power to its corresponding ESF Division. Therefore, a single failure within any system or within the electrical distribution systems will not prevent safe shutdown of the plant.

(continued)

BASES

LCO (continued)

OPERABILITY is met, as it applies to AC And DC Distribution Systems, provided the associated bus is energized to its proper voltage. The AC vital bus is OPERABLE when it is powered from its associated inverter and DC bus at proper voltage [and frequency]. Supplying the AC vital buses from the Class 1E constant voltage source is not acceptable for considering the AC Power Distribution System Division to be OPERABLE except as allowed by the Note in the LCO. In addition, Class 1E Divisional distribution cross-tie breakers open between redundant buses is required for distribution system OPERABILITY.

Inoperable distribution systems do not necessarily result in inoperable components unless directed by Required Actions.

This LCO is modified by a note allowing two inverters to be disconnected from their associated DC buses for ≤24 hours. This allowance is provided to perform an equalizing charge on the associated battery banks. During an equalizing charge, the resulting voltage condition may damage equipment energized from the associated vital buses. Disconnecting the two inverters is allowed provided the associated vital buses are energized from their Class 1E constant voltage source transformer and the AC vital buses for other battery banks are energized from the associated inverters connected to their DC buses. These provisions minimize the loss of equipment that would occur in the event of a loss of offsite power. The 24-hour time period for the allowance minimizes the time during which a loss of offsite power could result in the loss of equipment energized from the affected AC vital bus while taking into consideration the time required to perform an equalizing charge on the battery bank. When utilizing the allowance, if one or more of the provisions is not met (e.g., 24-hour time period exceeded, etc.), the associated vital bus must be declared inoperable.

APPLICABILITY

The AC and DC Power Distribution Systems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

1. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of anticipated operational occurrences or abnormal transients, and

(continued)

BASES

APPLICABILITY (continued)

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

AC and DC power distribution system requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, Distribution Systems - Shutdown.

ACTIONS

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

With one or more required buses, except AC vital buses, in one division inoperable, the remaining Power Distribution System Division is capable of supporting the minimum safety functions necessary to shutdown the unit and maintain it in a safe shutdown condition, assuming no single failure. However, the overall reliability is reduced because a single failure in the remaining division of the Power Distribution System could result in the minimum required ESF functions not being supported. Therefore, the required DC buses must be restored to OPERABLE status within two hours and the AC buses must be restored to OPERABLE status within eight hours. The two-hour Completion Time for DC buses is consistent with Reference 2. The eight-hour Completion Time is based on industry-accepted practice and engineering judgment taking into consideration the number of available systems and the time required to reasonably complete the Required Actions.

The specific list of features encompassed by Required Action A.2 is provided in Reference 2. These features are those which are designed with redundant safety related divisions. Single division systems are not included. Since the Completion Time allowance for this Required Action is limited to two hours, those systems with allowed Completion Times ≥ two hours for both division inoperable are not included as required features to be checked. Required Action A.2 is intended to provide assurance that a loss of offsite power, during the period that an AC bus is inoperable, will not result in a complete loss of safety function of a critical system. The term "ensure," as used in Required Action A.2, allows for an administrative check by examining logs or other information to determine if certain features are out-of-service for maintenance or other reasons. It does not require unique performance of the Surveillance Requirements needed to demonstrate OPERABILITY of the feature. The Completion Time is intended to allow the operator time to

(continued)

BASES

ACTIONS (continued)

A.1, A.2, and A.3 (continued)

evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both (1) an inoperable AC bus exists, and (2) a required feature on the other division is inoperable. If at any time during the existence of this Condition (one AC bus is inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked. The Completion Time is based on engineering judgment taking into consideration the probability of a loss of offsite power occurring in this condition, thereby resulting in a loss of function of this feature. This is comparable to both diesel generators inoperable and, therefore, a comparable Completion Time is established.

B.1 and B.2

With one AC vital bus inoperable, the remaining OPERABLE AC vital buses are capable of supporting the minimum safety functions necessary to shutdown the unit and maintain it in the safe shutdown condition. However, overall reliability is reduced since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the required AC vital bus must be powered from its alternate Class 1E constant voltage source transformer within two hours. Within 24 hours, the AC vital bus must be restored to OPERABLE status. This ensures that equipment powered from the AC vital bus will remain OPERABLE during a loss of offsite power. The 24-hour Completion Time is based on engineering judgment taking into consideration the probability of a loss of offsite power occurring during this time period, the number of available systems, and the time required to reasonably complete the Required Actions.

C.1 and C.2

The plant must be placed in a MODE in which the LCO does not apply if the Required Actions and associated Completion Times are not met. This is done by placing the plant in at least MODE 3 in six hours and in MODE 5 in 36 hours. The allowed Completion Times are reasonable based on operating experience to reach the required MODES from full power without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This surveillance verifies that the AC and DC Power Distribution Systems are functioning properly with all the desired circuit breakers closed and the buses energized from normal power. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The seven-day Frequency is based on engineering judgment and industry accepted practice considering the unit conditions required to perform the test, the ease of performing the test, and the likelihood of a change in the system component status.

SR 3.8.7.2

This surveillance verifies that the inverters are functioning properly, ensuring that the frequency on the AC vital buses are within limits. The seven-day Frequency is based on engineering judgment considering the unit conditions required to perform the test, the ease of performing the test, and the likelihood of a change in the system component status.

REFERENCES

1. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.
2. Regulatory Guide 1.93, "Availability of Electric Power Sources", December 1974.

BASES

Table B 3.8.7-1

POWER DISTRIBUTION SYSTEM

TYPE	VOLTAGE	DIVISION 1	DIVISION 2
AC Emergency Buses	4160 VAC	[ ]	[ ]
	480 VAC	[ ]	[ ]
DC Buses	125 VDC	[ ] from battery	[ ] from battery
		[ ] from charger [ ]	[ ] from charger [ ]
		[ ] from battery	[ ] from battery
		[ ] from charger [ ]	[ ] from charger [ ]
AC Vital Buses	120 VAC	[ ] from inverter	[ ] from inverter
		[ ] from inverter [ ]	[ ] from inverter [ ]

16A.11.8 B 3.8.8 DISTRIBUTION SYSTEMS - SHUTDOWN

B 3.8 ELECTRICAL POWER SYSTEMS

B.3.8.8 Distribution Systems - Shutdown

BASES

BACKGROUND	A description of the AC and DC Power Distribution Systems is provided in the Bases for LCO 3.8.7, Distribution Systems - Operating.
APPLICABLE SAFETY ANALYSES	In MODES 1, 2, 3, and 4, the requirement to have two AC and DC Power Distribution Systems OPERABLE is consistent with the initial assumptions of the accident analyses and the design requirements and are based upon maintaining at least one of the AC and the corresponding DC power sources and associated distribution systems OPERABLE during accident conditions in the event of (1) an assumed loss of all offsite or all onsite AC power, and (2) a single failure of the other AC or DC source. In MODES 5 and 6, the design basis accident (DBA) assumptions of a loss of coolant accident (LOCA) coincident with a loss of offsite or onsite AC power and a single failure are not required to be met. Therefore, the requirement to have one AC and DC Power Distribution System OPERABLE is adequate to assure power is available in MODES 5 and 6 and when handling irradiated fuel to systems required to recover from an inadvertent draindown of the reactor vessel or a fuel handling accident (Ref. 1). The OPERABILITY of the minimum specified AC and DC power sources and associated distribution systems during shutdown and refueling ensures that (1) the facility can be maintained in the shutdown or refueling condition for extended time periods, and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status (Ref. 2). The CESSAR-DC only addresses bounding analyses, such that a specific design basis is not always stated for operation in MODES 5 and 6. The safety analysis assumptions for a design basis event may not be applicable to operation in MODES 5 and 6. Distribution Systems - Shutdown satisfies the requirements of Selection Criteria 3 of the NRC Interim Policy Statement as documented in Reference 2.

(continued)

BASES

LCO

In MODES 5 and 6 and when handling irradiated fuel, one of the AC and DC Power Distribution System Divisions is required to be OPERABLE. This ensures the availability of sufficient power to recover from postulated events in MODES 5 and 6 and when handling irradiated fuel (e.g., fuel handling accident, reactor vessel draindown).

Inoperable distribution systems do not necessarily result in inoperable components unless directed by Required Actions (refer to LCO 3.0.7). For components powered from buses not listed in Table B 3.8.7-1, LCO 3.0.7 does not allow an exception to the definition of OPERABILITY and any of these buses deenergized would result in all affected components inoperable.

APPLICABILITY

The AC and DC Power Distribution Systems Divisions required to be OPERABLE in MODES 5 and 6 and when handling irradiated fuel assures sufficient power to ensure that:

1. Adequate coolant inventory makeup is available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel,

2. Systems needed to mitigate a fuel handling accident are available, and

3. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown or refueling condition.

AC and DC Power Distribution Systems requirements for MODES 1, 2, 3, and 4 are addressed in LCO 3.8.7, Distribution Systems - Operating.

ACTIONS

A.1, A.2, A.3, A.4, and A.5

With less than the required AC or DC Power Distribution Systems Division OPERABLE, no AC and/or DC power is available to recover from postulated events in MODES 5 and 6 and when handling irradiated fuel.

(continued)

BASES

ACTIONS (continued)

A.1, A.2, A.3, A.4, and A.5 (continued)

If applicable, CORE ALTERATIONS and handling of irradiated fuel must be immediately suspended. 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 operations with a potential to drain the reactor vessel and subsequent potential for fission product release. Also, if applicable, actions must be initiated immediately to suspend any operations involving positive reactivity additions. Actions must continue until the operations are suspended. These actions preclude the occurrence of the postulated events.

The requirement to immediately initiate action to restore the required AC and DC Power Distribution Systems to OPERABLE status is to be continued until restoration is completed. This is to minimize the time the plant is in this degraded condition of potentially (or actually) not being capable of recovering from the postulated fuel handling accident or reactor vessel draindown event.

The Completion Times are based on engineering judgment taking into consideration that a time of as short a duration as is practical should be specified while ensuring that the activities are suspended in a controlled manner.

SURVEILLANCE REQUIREMENTS

The Bases provided for SR 3.8.7.1 and SR 3.8.7.2, in the Bases for LCO 3.8.7, Distribution Systems - Operating, are applicable.

REFERENCES

1. Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors, NUREG-0212, Revision 3, December 1981.

2. NRR Memorandum, Dennis M. Crutchfield to Distribution, Subject: Technical Specifications OPERABILITY Requirements, dated July 8, 1985.

3. CESSAR-DC, Chapter 15.

(continued)

BASES

REFERENCES (continued)

4. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 8, 1987.

16A.12 B 3.9 REFUELING OPERATIONS

16A.12.1 B 3.9.1 BORON CONCENTRATION

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentration of the Reactor Coolant System (RCS), refueling cavity and refueling canal during refueling ensures that the reactor remains subcritical during MODE 6. The limit includes an uncertainty allowance of 50~~gpm~~.
ppm.

Refueling boron concentration is the solu. .e boron concentration in the reactor coolant during refueling or fuel handling. The soluble boron concentration offsets the fuel reactivity and is measured by chemical analysis of the reactor coolant. The refueling boron concentration specified in the COLR maintains overall core reactivity $\leq 0.95\ K_{eff}$ during fuel handling with Control Element Assemblies (CEAs) and fuel assemblies assumed to be in the most adverse (least negative reactivity) configuration allowed by plant procedures.

General Design Criteria 26 of 10 CFR Part 50, Appendix A requires two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The plant is brought to shutdown conditions before beginning operations to open the RCS for refueling. After the plant is cooled and depressurized, and the reactor vessel head is unbolted, the head is slowly raised. The refueling cavity and canal are then flooded by pumping borated water from the In-containment Refueling Water Storage Tank (IRWST) using the Containment Spray ~~(CS).~~
System pump(s).

If additions of boron are required after the vessel has been opened, the CVCS makes the additions through the RCS and open vessel. The pumping action of the SCS ~~SDC system~~ and natural circulation due to thermal driving heads in the vessel and cavity mix the added concentrated boric acid with the water in the

(continued)

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BASES

BACKGROUND (continued)

RCS and the refueling canal. The ~~SDC system~~ SCS is kept in service during the refueling period to remove core decay heat and provide forced circulation in the RCS.

APPLICABLE SAFETY ANALYSIS

During refueling operations the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The magnitude of the boron concentration is based on the nuclear design of each fuel cycle. It also guarantees that the K_{eff} of the core will remain less than 0.95 during the refueling operation.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling cavity, the refueling canal and the reactor vessel form a single mass. As a result, the soluble boron concentration is the same in each (Ref. 2).

The limiting boron dilution accident occurs in MODE 5, Reduced RCS Inventory ~~(Ref. 3)~~. A detailed discussion of this event is provided in ~~Bases 3.1.2, Shutdown Margin - T_{avg} ≤ 200°F.~~ Reference 6.

LCO

The LCO 3.9.1 requires that a minimum boron concentration be maintained while in MODE 6. The boron concentration limit during fuel handling operations ensures a K_{eff} of ≤ 0.95 is maintained. Violation of the LCO could lead to possible inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a K_{eff} of ≤0.95. Above MODE 6, LCO 3.1.1 ~~and 3.1.2,~~ Shutdown Margin, ensure that an adequate amount of negative reactivity is available to shutdown the reactor and to maintain the reactor subcritical.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions is contingent upon maintaining the plant in compliance with the LCO. With the

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

boron concentration of any of the filled portions of the RCS, the refueling canal, or the refueling cavity less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of actions to establish a safe condition.

A.3

In addition to A.1 and A.2, boration to restore the concentration must be initiated. The 15-minute completion time is the time allowed for an operator to correctly align and start the required systems and components.

In the determination of the required combination of boration flow rate and boron concentration, there is not a unique design basis event which must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration of the RCS as soon as possible, the boration solution should be a highly concentrated solution of boric acid.

Once boration is initiated, it must be continued until the boron concentration is restored. The completion time depends on the amount of boron which must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR verifies the reactor coolant boron concentration in the RCS, refueling canal and refueling cavity is within the COLR limit. The boron concentration in the coolant is determined periodically by chemical analysis.

Because the likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote, a minimum frequency of once every 72 hours is a reasonable interval to verify boron concentration. The surveillance interval is based on extensive operating experience and ensures that the boron concentration is checked at adequate intervals.

(continued)

BASES

REFERENCES

1. 10 CFR 50, Appendix A, Section VI, Criterion 26, "Reactivity Control System Redundancy and Capability."

2. NS-51.2, ANSI/ANS-57.2-1983, Section 6.4.2.2.3, American Nuclear Society, American National Standard, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants," 1983.

3. CESSAR-DC Chapter 15, Accident Analysis.

4. 52 FR 3788, NRC Interim Policy Statement, on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

Additional References

5. NRC Bulletin No. 89-03, "Potential Loss of Required Shutdown Margin During Refueling Operations," November 21, 1989.

6. CESSAR-DC, Section 15.8 "Shutdown Risk Report"

16A.12.2 B 3.9.2 NUCLEAR INSTRUMENTATION

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND	The installed source range monitors are part of the Nuclear Instrumentation System (NIS). These detectors are external to the reactor vessel and detect neutrons leaking from the core. The use of portable detectors is permitted, provided the LCO requirements are met. The installed source range monitors are BF3 detectors operating in the proportional region of the gas-filled detector characteristic curve. They monitor the neutron flux in counts per second (cps) and cover 5 decades of neutron flux (1 to 1E5 cps). Each source range monitor provides visual indication in the control room. If used, portable detectors should be functionally equivalent to the installed NIS source range monitors.
APPLICABLE SAFETY ANALYSES	OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to changes in core reactivity such as a boron dilution accident or an improperly loaded fuel assembly.
LCO	The OPERABILITY of two source range neutron flux monitors with visual indication in the control room ensures that redundant monitoring capability is available to detect changes in core reactivity.
APPLICABILITY	In MODE 6 the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. No other direct means are available. In MODES 2, 3, 4, and 5 the installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, Reactor Trip System Instrumentation.

(continued)

BASES

ACTIONS

A.1 and A.2

With one source range neutron flux monitor inoperable, redundancy has been lost. Since these instruments provide the only direct means of monitoring core reactivity conditions, CORE ALTERNATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of actions to establish a safe condition.

A.3

With one source range monitor inoperable, action shall be initiated to restore the inoperable monitor to OPERABLE status within seven days. Seven days is a reasonable time in which corrective actions must be initiated considering the boron sampling frequency of SR 3.9.1.1 and the suspension of CORE ALTERATIONS and positive reactivity changes per Required Actions A.1 and A.2 above. The seven days allows for instrument calibration and maintenance. Corrective actions, once started, must be continued until the monitor is restored to OPERABLE status.

B.1

With no source range monitor OPERABLE, actions to restore a monitor to OPERABLE status shall be initiated within 15 minutes. Once initiated actions shall be continued until a source range monitor is restored to OPERABLE status. With the unit in MODE 6 and all CORE ALTERATIONS and positive reactivity additions suspended, 15 minutes is a reasonable time to initiate corrective actions.

B.2

With no source range monitor OPERABLE, there is no direct means of detecting changes in core reactivity.

Performing SR 3.9.1.1 verifies that the required boron concentration exists. The four-hour completion time quickly verifies the boron concentration of the reactor coolant. The frequency of once per 12 hours ensures that unplanned changes in boron concentration would be identified. The 12-hour frequency is reasonable considering the low probability of a change in core reactivity during this time period.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is the comparison of the indicated parameter values for each of the functions. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels but each channel should be consistent with its local conditions. The frequency of 12 hours is based on the importance of the source range neutron flux monitors as the frequency is consistent with LCO 3.3.10, Engineered Safety Features Actuation System Instrumentation and has been proven acceptable through operating experience.

SR 3.9.2.2

The CHANNEL FUNCTIONAL TEST, performed once per seven days, ensures systematic verification of the OPERABILITY of the source range monitors. The frequency of seven days is based on the importance of the source range monitors as the only direct means of monitoring core reactivity and has been shown through operating experience to be a conservative interval considering operating history data for the setpoint drift.

REFERENCES

None.

16A.12.3 B 3.9.3 CONTAINMENT PENETRATIONS

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel within containment, a release of fission product radioactivity within the containment will be restricted from leakage to the environment when the LCO requirements are met. In MODES 1, 2, 3 and 4 this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1. In MODE 6 the potential for containment pressurization as a result of an accident is not present, therefore, less stringent requirements are needed to isolate the containment from the outside atmosphere.

The containment structure serves to contain fission product radioactivity which may be released from the reactor core following a Design Basis Accident (DBA), such that offsite radiation exposures are maintained within the requirements of 10 CFR 100. Additionally, this structure provides radiation shielding from the fission products which may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of fuel assemblies within containment with irradiated fuel in containment, the equipment hatch must be held in place by at least 4 bolts. Good engineering practice dictates that these four bolts be approximately equally spaced.

The containment personnel locks, which are also part of the containment pressure boundary, provide a means for personnel access during plant operation. Each personnel lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment closure is required.

During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an airlock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of fuel

(continued)

BASES

BACKGROUND (continued)

assemblies within containment with irradiated fuel in the containment, the door interlock mechanism may remain disabled, but one personnel lock door must remain closed.

The requirements on containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from leaking to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge and Exhaust System was designed for intermittent operation and provides a means of removing airborne radioactivity caused by minor operational Reactor Coolant System (RCS) leakage prior to personnel entry into containment. The system is also used for reducing the humidity and temperature inside containment. The system consists of four lines penetrating containment, two for supply and two for exhaust. The lines are sized to sufficiently reduce the airborne radioactivity level within containment to levels defined in 10 CFR 20 (Ref. 1) for a 40 hour work week within two hours of purge initiation.

During fuel movement the four purge and exhaust penetrations must be isolated or isolable and the other penetrations which provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by OPERABLE automatic containment purge or exhaust isolation valves or by an isolation valve, blind flange, or equivalent. Equivalent isolation methods may include a closed system within containment, or a material which can provide a temporary, atmospheric pressure, ventilation barrier for the

Containment penetrations during fuel movements. An example of such a material which has been approved for this use is a silicone foam sealant (Ref. 2).

OPERABILITY of the Containment Purge and Exhaust Isolation System, in accordance with LCO 3.6.3 Containment Isolation Valves, ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radioactivity level within containment. OPERABILITY of this system is required to restrict the release of radioactive material from containment atmosphere to the environment.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS

During CORE ALTERATIONS or movement of fuel assemblies within containment with irradiated fuel in containment, the most severe radiological consequences results from a fuel handling accident. The fuel handling accident is a Condition IV postulated event which involves damage to irradiated fuel (Ref. 3). Fuel handling accidents, analyzed in CESSAR-DC Section 15.7.3.4, include dropping a single fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies (Ref. 4). The requirements of this LCO and LCO 3.9.6 and the minimum decay time of 72 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity subsequent to a fuel handling accident does not result in doses in excess of the guideline values specified in 10 CFR 100 and Standard Review Plan Section 15.7.4, Rev. 1 (Ref. 3).

LCO

This LCO minimizes the consequences of a fuel handling accident in containment by minimizing the release of fission product radioactivity from the containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the containment purge and exhaust penetrations.

For the containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of fuel assemblies with irradiated fuel in containment since this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3 and 4, Containment Penetration requirements are addressed by LCO 3.6.1, Containment. In MODES 5 and 6 when CORE ALTERATIONS or movement of fuel assemblies in containment with irradiated fuel in containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

With the containment equipment hatch, personnel locks, or any containment penetration providing direct access from the containment atmosphere to the

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

outside atmosphere not in the required status, including the Containment Purge and Exhaust system not capable of automatic actuation when the Purge and Exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of fuel assemblies within containment. Performance of Required Action A.1 shall not preclude completion of actions to establish a safe condition.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This SR verifies that each of the containment penetrations, required to be in its closed position is in that position or is capable of being closed by an OPERABLE automatic Containment Purge and Exhaust Isolation System. As such, this surveillance ensures that a postulated fuel handling accident which involves a release of fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

The SR is performed every 12 hours during CORE ALTERATIONS or movement of fuel assemblies in the containment with irradiated fuel in containment. The surveillance interval is based on the importance of these penetrations to restrict the release of fission product radioactivity to the environment and has been shown to be acceptable through operating experience.

SR 3.9.3.2

This SR demonstrates each Containment Purge and Exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated Containment Isolation Signal, High Radiation Signal and High Relative Humidity Signal. The 18-month frequency maintains consistency with similar ESFAS testing requirements and has been shown to be acceptable through operating experience.

(continued)

BASES

REFERENCES

1. 10 CFR 20, Standards For Protection Against Radiation.

2. "Use of Silicone Sealant to Maintain Containment Integrity - ITS", GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.

3. NUREG-0800, Standard Review Plan Section 15.7.4, Radiological Consequences of Fuel Handling Accidents, Rev. 1, July 1981.

4. CESSAR-DC Section 15.7.3.4, Design Basis Fuel Handling Accidents.

16A.12.4 B 3.9.4 SHUTDOWN COOLING SYSTEM AND COOLANT CIRCULATION - HIGH WATER LEVEL

B 3.9 REFUELING OPERATIONS

B 3.9.4 Shutdown Cooling System and Coolant Circulation - High Water Level

BASES

BACKGROUND

The two main purposes of the Shutdown Cooling ~~(SDC)~~ System (SCS) are to remove decay heat and sensible heat from the Reactor Coolant System (RCS) when RCS pressure and temperature are below approximately 350 psig and 350°F, respectively (Ref. 1), and provide sufficient coolant circulation to minimize the effects of a boron dilution accident and prevent boron stratification. Heat is transferred from the RCS by circulating reactor coolant through the ~~SDC System~~ SCS where the heat is transferred to the Component Cooling Water (CCW) System via the ~~SDC~~ SCS heat exchangers.

In the decay heat removal mode of operation, each loop of the ~~SDC System~~ SCS takes suction from one of the RCS hot legs. Flow from the ~~Shutdown Cooling (SDC)~~ SCS pumps is discharged through its respective heat exchanger or bypass, and is returned to the RCS via the RCS cold legs. This arrangement provides two redundant ~~SDC~~ SCS loops. Operation of the ~~SDC System~~ SCS for normal cooldown or decay heat removal is manually accomplished from the control room.

APPLICABLE SAFETY ANALYSES

With the plant in MODE 6, the ~~SDC System~~ SCS is not required to mitigate any events or accidents evaluated in the safety analyses (Ref. 2). The NRC Interim Policy Statement requires that the ~~SDC System~~ SCS in MODE 6 be retained in the Technical Specifications. None of the selection criteria of the Interim Policy Statement were satisfied; however, it is the Commissions's policy that licensees retain in their Technical Specifications systems which operating experience and probabilistic risk assessment have generally shown to be important to public health and safety (Ref. 3).

LCO

Only one ~~SDC~~ SCS loop is required for decay heat removal in MODE 6 with water level ≥ ~~24~~ 23 feet above the top of the reactor vessel flange. Only one ~~SDC~~ SCS loop is required because the volume of water above the reactor vessel flange provides backup decay heat capability. At least one ~~SDC~~ SCS loop must be OPERABLE and in operation to:

(continued)

BASES

LCO (continued)

a. Provide for decay heat removal,

b. Provide mixing of borated coolant to minimize the possibility of a criticality, and

c. Provide indication of average reactor coolant temperature.

An OPERABLE loop consists of an available pump and flow path with adequate heat removal capability for existing plant conditions.

The requirements of this LCO are derived primarily from experience with decay heat removal in shutdown modes of operation. The principal purpose of this specification is to assure the capability to remove decay heat and to control RCS temperature and chemistry.

The LCO is modified by a Note which allows the operating ~~SDC~~ SCS loop to be removed from service for up to one hour per two-hour period provided no operation that would cause dilution of the RCS boron concentration is in progress. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS-to-~~SDC~~ SCS isolation valve testing. During this one-hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

One SDC loop must be OPERABLE and in operation in MODE 6 with the water level ≥ ~~24~~ 23 feet above the top of the reactor vessel flange to provide decay heat removal. The ~~24~~ 23 foot value was selected because it corresponds to the ~~24~~ 23 foot requirement established for fuel movement established by LCO 3.9.6. Requirements for the ~~SDC System~~ SCS in other MODES are covered by LCOs in Chapter 3.4, Reactor Coolant System.

~~SDC~~ SCS loop requirements in MODE 6 when water level is < ~~24~~ 23 feet are located in LCO 3.9.5, SDC - Low Water Level.

SCS loop requirements in Reduced RCS Inventory are addressed in LCO 3.10.4, Reduced RCS Inventory - Heat Removal

(continued)

BASES

ACTIONS

~~SDC~~ SCS loop requirements are met by having one ~~SDC~~ SCS loop OPERABLE and in operation except as permitted in the note to the LCO.

A.1

With ~~SDC~~ SCS loop requirements not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur by the addition of water with lower boron concentration than that contained in the RCS. Therefore, actions which reduce boron concentration shall be suspended immediately.

A.2

With ~~SDC~~ SCS loop requirements not met, actions shall be taken immediately to suspend operations involving an increase in reactor decay heat load. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of ~~24~~ 23 feet above the reactor vessel flange provides an adequate available heat sink. Suspending any operation which would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.3

With ~~SDC~~ SCS loop requirements not met, actions shall be taken and continued to satisfy the ~~SDC~~ SCS loop requirements. With the unit in MODE 6 and the refueling cavity water level ≥ ~~24~~ 23 feet above the top of the reactor vessel flange, 15 minutes is a reasonable time to initiate corrective actions.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This surveillance verifies that the ~~SDC~~ SCS loop is operating and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The frequency of 12 hours is sufficient considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the ~~SDC System~~ SCS in the control room. This frequency ensures that ~~SDC~~ SCS loop operation and flow is checked at adequate intervals.

(continued)

BASES

REFERENCES

1. CESSAR-DC Section 5.4.7, Shutdown Cooling System.

2. CESSAR-DC Chapter 15, Accident Analysis.

3. "NRC Staff Review of Nuclear Steam Supply Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," transmitted by Thomas E. Murley (NRC) letter to Joseph K. Gasper (CEOG) dated May 9, 1988.

4. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

16A.12.5 B 3.9.5 SHUTDOWN COOLING SYSTEM AND COOLANT CIRCULATION - LOW WATER LEVEL

B 3.9 REFUELING OPERATIONS

B 3.9.5 Shutdown Cooling System (SCS) and Coolant Circulation - Low Water Level

BASES

BACKGROUND

The Background section for Bases B 3.9.4 is applicable to this Bases.

APPLICABLE SAFETY ANALYSES

With the plant in MODE 6, the ~~RHR System~~ SCS is not required to mitigate any events or accidents evaluated in the safety analyses (Ref. 2). The NRC Interim Policy Statement requires that the ~~RHR System~~ SCS in MODE 6 be retained in the Technical Specifications. None of the selection criteria of the Interim Policy Statement were satisfied; however, it is the Commissions's policy that licensees retain in their Technical Specifications systems which operating experience and probabilistic risk assessment have generally shown to be important to public health and safety (Ref. 3).

LCO

Only one ~~SDC~~ SCS loop is required for decay heat removal in MODE 6 with water level < 24 feet above the top of the reactor vessel flange. To increase reliability, both ~~SDC~~ SCS loops must be OPERABLE. Additionally, one loop of SDC must be in operation in order to:

a. provide for decay heat removal,

b. provide mixing of borated coolant to minimize the possibility of a criticality, and

c. provide indication of average reactor coolant temperature.

An OPERABLE loop consists of an available pump and flow path with adequate heat removal capability for existing plant conditions.

The requirements of this LCO are derived primarily from experience with decay heat removal in shutdown modes of operation. The principal purpose of this specification is to assure the capability to remove decay heat and to control RCS, temperature, and chemistry with low water level.

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BASES

APPLICABILITY

Two ~~SDC~~ SCS loops are required to be OPERABLE and one ~~SDC~~ SCS loop must be in operation in MODE 6 with the water < ~~24~~ 23 feet above the top of the reactor vessel flange to provide decay heat removal. Requirements for the ~~SDC System~~ SCS in other MODES are covered by LCOs in Chapter 3.4, Reactor Coolant System. MODE 6 requirements with water level ≥ ~~24~~ 23 feet above the reactor vessel flange are covered in LCO 3.9.4, ~~SDC~~ SCS - High Water Level.

ACTIONS

A.1.1 and A.1.2

With one ~~SDC~~ SCS loop inoperable, actions shall be taken and continue until the ~~SDC~~ SCS loop is restored to OPERABLE status or to establish ≥ ~~24~~ 23 feet of water level is established above the reactor vessel flange where the Applicability will change to that of LCO 3.9.4 and only one ~~SDC~~ SCS loop is required OPERABLE and in operation. With the unit in MODE 6, 15 minutes is a reasonable time to initiate corrective actions.

A.2

Alternate decay heat removal capabilities shall be established to provide a backup method of decay heat removal in the event one of the OPERABLE ~~SDC~~ SCS loops becomes inoperable. Alternate decay heat removal methods are available to the operators for review and pre-planning in the unit Abnormal Procedures. This capability may be established by aligning other pumps and systems, such as containment spray pumps and heat exchanger or the charging pump through the CVCS, to provide reactor coolant circulation. A Completion Time of seven days limits the time a backup heat sink is not available. This is based on the reliability of an operating ~~SDC~~ SCS loop and has been shown to be acceptable through operating experience.

B.1

With no ~~SDC~~ SCS loop in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur by the addition of water with lower boron concentration than that contained in the RCS. Therefore, actions which reduce boron concentration shall be suspended immediately.

(continued)

BASES

ACTIONS (continued)

B.2.1 and B.2.2

With no ~~SDC~~ SCS loop in operation or with both ~~SDC~~ SCS loops inoperable, actions shall be initiated immediately and continued without interruption to restore one ~~SDC~~ SCS loop to OPERABLE status and operation. As the unit is in Conditions A and B concurrently, the restoration of two OPERABLE ~~SDC~~ SCS loops and one operating ~~SDC~~ SCS loop should be accomplished as quickly as possible. With at least one ~~SDC~~ SCS loop operable water level can be raised ≥ ~~24~~ feet above the reactor vessel flange when the applicability will change to that of LCO 3.9.4 and only one ~~SDC~~ SCS loop is required.

B.3

With no ~~SDC~~ SCS loop in operation or both ~~SDC~~ SCS loops inoperable, actions shall be initiated within 15 minutes to implement alternate decay heat removal as specified in plant procedures. Decay heat removal may be accomplished by use of the containment spray pumps and heat exchangers or the charging pumps through the CVCS with consideration for the boron concentration. The method used to remove decay heat should be the most prudent and safe choice based upon plant conditions. The choice could be different if the reactor vessel head is in place than if the reactor vessel head is removed. With the unit in MODE 6, 15 minutes is an adequate time to initiate action to implement alternate decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This surveillance verifies that the ~~SDC~~ SCS loop is operating and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal and to prevent thermal and boron stratification in the core.

In addition, during operation of the ~~SDC~~ SCS loop with the water level in the vicinity of the reactor vessel nozzles, the ~~SDC~~ SCS loop flow rate determination must also consider the ~~SDC~~ SCS pump suction requirements. The frequency of 12 hours is sufficient considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the ~~SDC~~ SCS system in the control room. This frequency ensures that flow is checked and temperature monitored at adequate intervals.

(continued)

BASES

REFERENCES

1. CESSAR-DC Section 5.4.7, Shutdown Cooling System.
2. CESSAR-DC Chapter 15, Accident Analysis.
3. "NRC Staff Review of Nuclear Steam Supply Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," transmitted by Thomas E. Murley (NRC) letter to Joseph K. Gasper (CEOG) dated May 9, 1988.
4. 52 FR 3788, NRC Interim Policy Statement, on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

16A.12.6 B 3.9.6 REFUELING WATER LEVEL

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Water Level

BASES

BACKGROUND

Requirements on water level in the containment, the refueling cavity, the refueling canal, the fuel transfer canal, and the spent fuel pool during refueling ensure that sufficient water depth is available to remove 99% of the iodine gap activity released by the postulated rupture of an irradiated fuel assembly in containment (Ref. 1). The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory. The movement of fuel assemblies within containment with irradiated fuel in containment requires a minimum water level of ~~24~~ 23 feet above the top of the reactor vessel flange which assures offsite doses remain < 25% of the 10 CFR 100 limits as required in Reference 5.

APPLICABLE SAFETY ANALYSES

During movement of fuel assemblies, the water level in the refueling cavity and refueling canal is an initial condition design parameter in the analysis of the fuel handling accident in containment postulated by NRC Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 feet (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released, from the gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel handling accident analysis inside containment is described in Reference 2.

LCO

A minimum refueling water level of ~~24~~ 23 feet above the irradiated fuel is required to ensure that the radiological consequences of a postulated fuel handling accident inside Containment are within acceptable limits.

APPLICABILITY

LCO 3.9.6, Refueling Water Level, is applicable when moving fuel assemblies within containment with irradiated fuel assemblies in containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident.

(continued)

BASES

ACTIONS

A.1

With a water level of less than ~~24~~ 23 feet above the top of the reactor vessel flange, all operations involving movement of fuel assemblies shall be suspended immediately to ensure a fuel handling accident cannot occur. The suspension of fuel movement shall not preclude completion of movement to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of ~~24~~ 23 feet above the top of the reactor vessel flange ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the reactor vessel flange, mitigates the consequences of a postulated fuel handling accident inside containment which results in damaged fuel rods (Ref. 2).

The 24-hour frequency ensures that the water is at the required level and monitors the level to detect any unplanned changes in water level. Due to the large volume of water and the normal procedural controls of valve positions, significant unplanned level changes are unlikely. The frequency of 24 hours has proven by operating experience to be adequate to determine and monitor water level.

REFERENCES

1. USNRC Regulatory Guide 1.25, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, March, 1982.

2. CESSAR-DC Chapter 15, Accident Analysis.

3. 52 FR 3788, "Proposed Policy Statement on Technical Specifications Improvements for Nuclear Power Plants", February 6, 1987.

4. 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance."

(continued)

BASES

REFERENCES (continued)

5. NUREG-0800, "Standard Review Plan", Section 15.7.4 Radiological Consequences of Fuel Handling Accidents, U.S. Nuclear Regulatory Commission.

16A.13 B 3.10 REDUCED RCS INVENTORY OPERATIONS

16A.13.1 B 3.10.1 REACTOR TRIP CIRCUIT BREAKERS

B 3.10 REDUCED RCS INVENTORY OPERATIONS

B 3.10.1 Reactor Trip Circuit Breakers

BASES

BACKGROUND	During reduced inventory operations the reactor is maintained in a subcritical condition. The requirement for the reactor trip circuit breakers to be open during reduced RCS inventory operations ensures that the reactor will not become critical due to the inadvertant withdrawal of Shutdown CEA's.
APPLICABLE SAFETY ANALYSIS	During reduced RCS inventory operations reactor trip circuit breakers are relied upon to mitigate events analyzed in Reference 1 which would result in a positive reactivity addition greater than the allowable limits identified in Reference 2. With the breakers open the CEA's cannot be withdrawn and reactivity accidents of this type are prevented.
LCO	LCO 3.10.1 requires that the reactor trip circuit breakers are open during reduced inventory operations to ensure that accidental reactivity additions by CEA withdrawal are prevented and that the radiological consequences of postulated reactivity accidents are within the established limits.

BASES

APPLICABILITY

This LCO is applicable in MODE 5 with reduced RCS inventory only to ensure that the reactor will not be critical due to the inadvertant withdrawal of the shutdown CEA's. This LCO is not necessary in MODE 6 due to plant arrangement that requires the CEDM power cables be disconnected.

ACTIONS

If any reactor trip circuit breakers are closed they are to be immediately opened to ensure that there can be no inadvertant CEA withdrawal.

SURVEILLANCE REQUIREMENTS

SR 3.10.1.1

Verification of the reactor trip circuit breakers being open ensures that the CEA's cannot be inadvertantly withdrawn.

The frequency of 12 hours is based on engineering judgment and is considered adequate since normal procedural controls would make the unauthorized or inadvertant closing of reactor trip circuit breakers unlikely.

REFERENCES

1. System 80+ Shutdown Risk Evaluation Report, July 31, 1992, Section 2.6.3

2. CESSAR-DC Chapter 16, Technical Specifications, LCO 3.1.1

3. System 80+ Shutdown Risk Evaluation Report, July 31, 1992, Section 7.2.1

16A.13.2 B 3.10.2 REDUCED RCS INVENTORY OPERATIONS - INSTRUMENTATION

B 3.10 REDUCED RCS INVENTORY OPERATIONS

B 3.10.2 Reduced RCS Invertory -Instrumentation

BASES

BACKGROUND

Inadequate instrumentation can be a factor in the loss of shutdown cooling (Reference 1). To ensure that shutdown cooling is maintained, the NRC recommendations (Reference 2) regarding instrumentation during reduced RCS inventory operations have been adopted.

The installed instrumentation is provided to monitor RCS level, RCS temperature and SCS performance. Visual indication and alarms of the above mentioned parameters are provided in the control room to inform the operator of the conditions necessary for proper decay heat removal. If used, temporary instrumentation should be functionally equivalent to the installed instrumentation.

Four independant and redundant sets of water level monitoring instruments provide indication of RCS water level during MODE 5 Reduced RCS Inventory Operations. Level monitoring capability is available from the reduced inventory setpoint (3 feet below the reactor vessel flange) to the bottom of the hot leg. These level indicators are

(continued)

BASES

BACKGROUND (continued)

calibrated for low RCS temperature operation and are highly accurate.

The following is a discussion of the level instrumentation provided:

1. A pair of wide range dp based water level sensors provide indication of RCS water level from the pressurizer to below the minimum level required for SCS operation. Each level sensor measures RCS level from the reference tap located at the top of the pressurizer to a tap located at the hot leg/SCS suction line interface. Each sensor is independant and redundant.

2. A pair of narrow range dp based water level sensors provide indication of RCS water level during drain down operations. Level is monitored from reference leg taps located at the DVI nozzles to taps located at the hot leg/SCS suction line interfaces.

3. Two inadequate core cooling heated junction thermocouples (HJTC's) provide indication of RCS level from the reactor vessel head to the fuel alignment plate.

4. Two refueling water level HJTC instruments provide indication of RCS water level from the reactor vessel head to the fuel alignment plate with improved level detection capability across the hot leg region via clustered thermocouples located in that

(continued)

BASES

BACKGROUND (continued)

region.

Reduced inventory operations level sensors available for MODE 6 operations are provided as described in Items 1 and 2 above.

Instruments are available for continuous temperature measurements during reduced inventory operation. The following is a discussion of the temperature instrumentation provided:

1. RTD's are located in each hot leg near the SCS suction lines. The RTD's are available to monitor RCS temperature during reduced inventory operation in MODE 5 and MODE 6. The RTD's are an accurate measurement of reactor coolant temperature exiting the core when the SCS is in operation.

2. RTD's are located in the shutdown cooling system piping which provide RCS temperature indication in MODE 5 and MODE 6 when the SCS is operational. These RTD's are located at either shutdown cooling heat exchanger inlet and return line.

3. The CET's are capable of monitoring RCS temperature as it exits the core during normal or reduced RCS inventory levels with the reactor head installed. When the reactor vessel head is removed for refueling purposes, the RCS and SCS RTD's provide RCS temperature indication due to the unavailability of the CET's.

(continued)

BACKGROUND (continued)

4. The HJTC probes are capable of providing a continuous measurement of coolant temperature inside the vessel. The HJTC temperature sensors are available at normal or reduced RCS inventory levels when the reactor vessel head is installed. When the reactor vessel head is removed the RCS RTD's, SCS RTD's and CET's if available, provide RCS temperature indication.

In the event of a loss of shutdown cooling, the RTD's can no longer be relied upon to provide an exact indication of reactor coolant temperature exiting the core due to their location near the SCS suction lines in the hot legs and in the SCS piping.. In this situation, other instruments, such as the core exit thermocouples (CET's) and the HJTC probes, are available to monitor RCS temperature as it exits the core.

Sufficient information will be available to the control room operator to adequately monitor SCS performance. The following is a discussion of the SCS performance instrumentation provided:

1. SCS pump suction and discharge pressure transmitters are installed to provide control room indication of SCS pump operating pressures throughout the design pressure range. Alarms are provided in the control room to warn operators of a low suction or discharge pressure.

2. SCS flowrate is provided via an installed

(continued)

BASES

BACKGROUND (continued)

flowmeter in each SCS return line to the RCS. Alarms are provided in the control room to warn operators of a degraded SCS flow condition.

3. SCS/CS pump motor current is monitored using an ammeter that provides pump motor current in the control room. An alarm is provided to alert the operator of a preset drop in motor current.

4. SCS heat exchanger inlet and return line temperature sensors provide indication in the control room of SCS temperature. A heat exchanger Δt can demonstrate adequate SCS heat removal to verify support system heat removal capability.

5. SCS valve position indication is provided in the control room to alert the operator of system lineup and available SCS flow paths. Open, closed and throttled indication of the major SCS inlet and return valves is provided.

Indications of sufficient pump suction pressure and possible vortexing include the following:

1. Unsteady pump motor current as indicated by SCS/CSS pump motor amps.

2. Low SCS pump suction pressure

3. Low SCS flowrate

(continued)

BASES

BACKGROUND (continued)

4. Increasing RCS level due to air/vapor displacement of water.

Vortex formation in the SCS suction line is a function of RCS water level and SCS flowrate. The higher the SCS flowrate, the hot leg level must be maintained to preclude vortexing. Requiring an adequate fluid level in the hot leg above the level at which vortexing occurs at the maximum allowable SCS flowrate will ensure that the suction line does not entrain air. Operations below 21 inches, hot leg centerline, are not recommended.

The instrumentation provided for monitoring RCS conditions affecting reduced inventory operations will significantly reduce risk associated with operating at a reduced inventory condition provided the instrumentation is placed in service and verified operable prior to the start of the draining evolution.

APPLICABLE SAFETY ANALYSIS

During reduced RCS inventory operations an accurate assessment is required of RCS conditions to both enhance monitoring capabilities for prevention of loss of shutdown cooling and provide for a timely response to a loss of shutdown cooling. The instrumentation covered in this LCO must be OPERABLE.

LCO

The LCO 3.10.2 requires a specified number of instruments be operable in order to closely monitor reduced RCS inventory operating parameters to

(continued)

LCO (continued)

assist in preventing a loss of core heat removal capabilities and ensure a timely response to a loss of shutdown cooling event.

The requirement of two independant means of monitoring RCS level ensures that continuous monitoring capability during RCS draindown is available. The level indicators provide indication from the pre-drain down normal level to a level below that necessary for SCS operation.

The requirement for one wide range and one narrow range level indicator ensures that sufficient instrumentation is available to cover draining from the pressurizer to the bottom of the hot leg and to accurately display the level within the hot leg once that level is reached.

The requirement of two independant means of monitoring RCS temperature ensures that continuous indication of temperature representative of core exit conditions is available regardless of reactor vessel head status.

The requirement of two independant indications available for monitoring SCS performance in the loop on service for decay heat removal ensures that sufficient instrumentation is available at all times to detect a degredation in SCS performance.

APPLICABILITY

This LCO is applicable in MODE 5 and MODE 6 with reduced RCS inventory conditions to ensure that the required instrumentation is provided to avoid

(continued)

APPLICABILITY (continued)

causing or contributing to a loss of shutdown cooling at reduced inventory conditions, to aid in correctly interpreting a loss of shutdown cooling, should one occur, and to assist in the restoration of decay heat removal, should a loss occur.

ACTIONS

A.1

In the event that wide range (WR) level instrumentation is declared INOPERABLE, immediate action must be taken to attempt to restore this indication. WR level indication is necessary to accurately determine reduced RCS inventory level at levels greater than the top of the hot leg.

While every attempt is being made to restore WR level indication, conditions such as RCS temperature, SCS performance and NR level shall be monitored and recorded every 30 minutes to determine if a trend is developing.

B.1

In the event that narrow range (NR) level instrumentation is declared INOPERABLE, immediate action must be taken to attempt to restore this indication. NR level indication is necessary to accurately determine reduced RCS inventory level at midloop elevations or elevations from the DVI nozzles to the bottom of the hot leg.

While every attempt is being made to restore

(continued)

BASES

ACTIONS (continued)

NR level indication, conditions such as RCS temperature and SCS performance shall be monitored and recorded every 30 minutes to determine if a trend is developing. The WR level indication shall be monitored and recorded every 10 minutes due to the loss of the preferred means of level indication.

B.2

In addition to ACTION B.1, it is necessary to take action to restore RCS level to greater than the reduced inventory elevation of [117'-0"] immediately. This is necessary because of the relatively small amount of coolant in the reactor vessel during reduced inventory, esp. mid-loop, boiling in the core could take place in as little as 15 to 20 minutes if the SCS were to become air bound due to an undetected reduction in RCS level.

C.1

In the event that RCS temperature monitoring has been reduced to a single means of temperature indication, immediate action shall be taken to restore RCS temperature monitoring to at least two independant indications.

During reduced RCS inventory operations several (as many as eight or more could be available at one time) independant temperature measurements representative of core exit temperature are provided. The availability of only one of these indications does not necessitate reduced inventory

(continued)

BASES

ACTIONS (continued)

recovery. Frequent monitoring [30 minutes] of the reduced inventory instrumentation described in this bases is sufficient to satisfy the LCO.

D.1 and D.2

In the event that RCS temperature monitoring capability is not available, immediate action shall be taken to restore RCS temperature monitoring to at least one indication. During the time period that RCS temperature indication is not available, reduced inventory instrumentation such as, SCS performance instrumentation and RCS level instrumentation, shall be monitored and recorded every 10 minutes.

Since temperature indication is valuable in determining decay heat removal adequacy, as well as guiding SCS restoration actions and monitoring the success of recovery actions, it is necessary to take action to restore RCS level to greater than the reduced inventory elevation of [117'-0"] immediately if temperature indication is not available.

E.1 and E.2

In the event that SCS performance monitoring capability is not available, immediate action shall be taken to restore SCS performance monitoring to at least two independant indications. During the time period that SCS performance indication is not available, reduced RCS inventory instrumentation such as, RCS temperature instrumentation and RCS

(continued)

BASES

ACTIONS (continued)

level instrumentation, shall be monitored and recorded every 10 minutes.

An additional choice is provided and should be followed to place an alternate (if applicable) decay heat removal system in operation. A two hour time period is stated to allow the operators to safely secure the running decay heat removal path and restart the alternate path should a mid-loop condition exist.

SURVEILLANCE REQUIREMENTS

SR 3.10.2.1

This surveillance requires verification of the WR and NR RCS level indication in service for monitoring reduced inventory level by performing a CHANNEL CHECK to determine the level indications are consistent with one another.

The frequency of 6 hours is based on the importance of RCS level indication during reduced inventory conditions and the ability to verify normal expected instrument drift.

The Data Processing System (DPS) continuously performs a cross channel comparison and will institute an alarm to warn operatoras that a channel has drifted out-of-tolerance or is not working properly. Should the DPS become inoperable the required CHANNEL CHECK shall be performed once every 6 hours (twice per shift) by the control room operators.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.2.2

This surveillance requires verification of the RCS temperature indications in service used for monitoring core exit/RCS temperature. This is accomplished by performing CHANNEL CHECK to determine the core exit/RCS temperature indications are consistent with one another.

The frequency of 6 hours is based on the importance of RCS temperature indication during reduced inventory conditions and the ability to verify normal expected instrument drift.

The Data Processing System (DPS) continuously performs a cross channel comparison and will institute an alarm to warn operatoras that a channel has drifted out-of-tolerance or is not working properly. Should the DPS become inoperable the required CHANNEL CHECK shall be performed once every 6 hours (twice per shift) by the control room operators.

SR 3.10.2.3

This surveillance requires verification of the SCS performance indicators in service used for monitoring decay heat removal capability. This is accomplished by performing CHANNEL CHECK to determine the SCS performance indicators are consistent with one another.

The frequency of 6 hours is based on the importance of SCS performance indicators during

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

reduced inventory conditions and the ability to verify normal expected instrument drift.

The Data Processing System (DPS) continuously performs a cross channel comparison and will institute an alarm to warn operatoras that a channel has drifted out-of-tolerance or is not working properly. Should the DPS become inoperable the required CHANNEL CHECK shall be performed once every 6 hours (twice per shift) by the control room operators.

SR 3.10.2.4

Performance of a CHANNEL CALIBRATION of the instruments in use for monitoring reduced RCS inventory parameters ensures that the channels have been recently calibrated and are reading accurately within specified tolerances prior to placing them in service and declaring them OPERABLE for reduced RCS inventory monitoring.

This calibration requirement of 60 days is intended to be more restrictive than the normal calibration frequencies specified for a particular instrument that may already be in service for other plant monitoring purposes.

REFERENCES

1. System 80+ Shutdown Risk Evaluation Report, July 31, 1992, Section 2.8

2. NUREG 1449

16A.13.3 B 3.10.3 REDUCED RCS INVENTORY OPERATIONS VENT PATHS

B 3.10 REDUCED RCS INVENTORY OPERATIONS

B 3.10.3 Reduced RCS Inventory Operations - Vent Paths

BASES

BACKGROUND	The requirement for a sufficient RCS vent path to be established during reduced RCS inventory operations prevents the pressurization of the RCS which would be anticipated upon inadequate DHR capability. This pressurization of the RCS could lead to SG nozzle dam failure and a potential loss of reactor coolant. When the pressurizer manway is opened to the containment atmosphere, it provides sufficient venting capacity to prevent core uncovery due solely to pressurization of the hot side resulting from boiling in the core coolant.
APPLICABLE SAFETY ANALYSIS	During reduced RCS inventory operations analyses were performed, Reference 1, and have indicated that with the pressurizer manway opened and relieving to the pressurizer cubicle, RCS boiling at reduced RCS inventory conditions will not cause a pressurization of the RCS that would exceed the SG nozzle dam design pressure of 40 psig.

BASES

LCO

The LCO 3.10.3 requires that during reduced RCS inventory operations a vent path of ≥ [Pressurizer Manway Removal] is established and maintained prior to and during reduced RCS inventory conditions. Other vent paths are acceptable provided analyses have been performed and results verify that pressure throughout the RCS is below the SG nozzle dam design pressure.

APPLICABILITY

This LCO is applicable in MODE 5 with reduced RCS inventory and MODE 6 reduced RCS inventory with the reactor vessel head in place and at least one reactor vessel stud tensioned.

When the reactor vessel head is removed, a sufficient vent path is established to prevent pressurization of the RCS and therefore does not apply during this condition.

ACTIONS

A.1, A.2 and A.3

Immediate action shall take place to restore the RCS vent path used for reduced inventory operations should it be discovered to be inoperable / isolated.

A time of [6 hours] is provided ***to allow for vent path restoration.*** ~~***and is based on the time required to pressurize the RCS to a value sufficient to cause SG nozzle dam failure.***~~

During the period of time that the vent path is

(continued)

BASES

ACTIONS (continued)

inoperable, reduced RCS inventory instrumentation such as RCS temperature, RCS level and SCS performance shall be monitored hourly by the control room operator in order to detect a trend leading to the loss of DHR.

B.1

The RCS level shall be restored to a level > reduced inventory elevation of [117'-0"] within [6 hours] should any of the allowed Completion Times mentioned above exceed the allotted time period. [6 hours] is considered reasonable time to secure RCS openings and restore RCS level.

SURVEILLANCE REQUIREMENTS

SR 3.10.3.1

Once the vent path is initially established it shall be verified established and unobstructed once per shift [12 hours] by operating personnel. Once per shift is considered a reasonable time interval for operating personnel to perform this verification.

REFERENCES

1. System 80+ Shutdown Risk Evaluation Report, July 31, 1992, Section 2.3.3.3

16A.13.4 B 3.10.4 RED. INVENTORY - HEAT REMOVAL

B 3.10 REDUCED RCS INVENTORY OPERATIONS

B 3.10.4 Reduced RCS Inventory Operations - Heat Removal

BASES

BACKGROUND

The shutdown cooling system (SCS) removes decay heat from the reactor coolant system and transfers the heat to the component cooling water (CCW) system.

During reduced RCS inventory operations the interruption or loss of SCS flow, DHR capability, can lead to bulk boiling and fuel uncovery quite rapidly. In some cases, as little as 15-20 minutes. During reduced RCS inventory operations, the SCS is the primary means of decay heat removal.

Each SCS division has a SCS pump, SCS heat exchanger, valves and connecting piping. In addition to these components, the containment spray system (CSS) pumps, which are are identical to the SCS pumps, with some minor valve manipulations can be used as a backup pumping source should the SCS pumps become inoperable.

APPLICABLE SAFETY ANALYSIS

During reduced RCS inventory operations the loss of one SCS division is within the design basis of the safety analyses described in Reference 1. Several means discussed in Reference 1 are available to provide alternate means of DHR should the primary means become inoperable.

BASES

LCO

The LCO 3.10.4 requires that two SCS divisions are OPERABLE and that one is in operation at all times and that the Containment Spray Pump in the operating SCS division is OPERABLE.

An OPERABLE division is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat removal cannot occur via the SCS unless forced flow is used.

The operability of the Containment Spray Pump in the operating SCS division is based on the ability to realign the affected SCS division to allow the CSS pump to perform DHR capabilities should a failure occur involving the SCS pump motor or electrical power.

This LCO does not permit periods of time where decay heat removal capability can be interrupted to permit surveillance testing or pump switching. It is understood that these activities could result in RCS boiling as discussed previously or may cause perturbations in RCS level which could lead to the air binding of the operating SCS division and subsequent RCS boiling.

APPLICABILITY

This LCO is applicable in MODE 5 and MODE 6 with reduced RCS inventory to ensure that decay heat is adequately removed from the RCS and that the necessary redundancy is provided during this condition.

BASES

ACTIONS

A.1 and B.1

With one SCS division inoperable, the division shall be declared OPERABLE within [15 minutes] or it will be necessary to raise RCS level to a height greater than elevation [117'-0"] to restore from reduced RCS inventory operations.

The time to RCS boiling calculations (Reference 1) can be as little as 15 minutes if decay heat removal is interrupted. The [15 minute] time limit is considered sufficient to allow the operator to determine the cause for the loss of redundancy and restore the division to OPERABLE status.

C.1, C.2, and C.3

If no SCS division is in operation then immediately suspend all operations involving the reduction in RCS boron concentration, initiate action to restore one SCS division to operable status and place it in operation, and initiate action to raise RCS level to > EL [117'-0"]. These actions are for the purpose of restoring core cooling and to prevent a boron dilution event.

D.1, D.2, and D.3

If the Containment Spray Pump in the operating SCS division is inoperable, action must be initiated to place the alternate division in operation (if the containment spray pump in the alternate division is

(continued)

BASES

ACTIONS (continued)

OPERABLE) within 6 hours. Also, SCS performance must be monitored every 30 minutes and the inoperable Containment Spray Pump must be restored to OPERABLE condition within 48 hours.

SURVEILLANCE REQUIREMENTS

SR 3.10.4.1

Verification of at least one SCS division in operation ensures that the plant is within the safety analysis. The interval of 12 hours (once per shift) is based on operating experience.

SR 3.10.4.2

Verification of the correct breaker alignment and indicated power available to the SCS pump that is not in operation and the operable CSS pump ensures that the alternate SCS division or redundant CSS pump will be able to remove heat from the RCS in the event of a power failure to the operating SCS division.

REFERENCES

1. CESSAR-DC, Section 5.4.7 "Shutdown Cooling System"

2. CESSAR-DC, Section 15.8, "Shutdown Risk Report"

B 3.10.5 Reduced RCS Inventory Operations - Containment Integrity

Background	During reduced RCS inventory operations, a release of fission product radioactivity within containment will be restricted from leakage to the environment when the LCO requirements are met.
Applicable Safety Analysis	Release of fission products to the environment from containment is limited by 10CFR100. If the LCO requirements are adhered to, then no release exceeding the 10CFR100 limits can occur (Ref. 1).
LCO	This LCO minimizes the release of radioactivity from containment. The LCO requires each penetration providing direct access to the outside environment to be closed with the exception of the containment purge and exhaust isolation system.
Applicability	The LCO is applicable during MODES 5 and 6 within the reduced RCS inventory conditions.
Actions	A.1 If one or more containment penetrations are not in the required status, restoration must be accomplished within 6 hours. This will ensure that the plant will be within the safety analysis. B.1 If Action A.1 has not been completed within the 6 hours then the RCS level must be restored to > EL-117' 0" within 6 hours of Action A.1 not being met.
Surveillance Requirements	SR 3.10.5.1 This SR verifies that each required containment building penetration is in its required status. This ensures that fission products will not escape containment in a quantity greater than the safety analysis. SR 3.10.5.2 This SR demonstrates each Containment Purge and Exhaust valve actuates to its isolated position on manual initiation or on an actual or simulated Containment Isolation Signal, High Radiation Signal and High Relative Humidity Signal. The 18-month frequency maintains consistency with similar ESFAS testing requirements and has been shown to be acceptable through operating experience.

Reference 1. System 80+ Shutdown Risk Evaluation Report, July 31, 1992, Section 2.5.

B 3.10.6 Reduced RCS Inventory Operations - AC Power Availability

Background	AC power must be available to a certain degree of reliability since decay heat removal capability must be maintained. AC power includes both the sources to the Class 1E distribution system and the diesel generators.
Applicable Safety Analysis	During reduced inventory operations, decay heat from the reactor must be removed. Electric power is required for operating the shutdown cooling system. The LCO requirements will maintain an adequate margin for the operability of the AC power sources (Ref. 1).
LCO	The LCO requires that two independent sources of AC power to each division supplying the Class 1E distribution system shall be operable. A diesel generator in either division must also be operable.
Applicability	LCO 3.10.6 is applicable during MODES 5 and 6 reduced RCS inventory operations.
Actions	A.1 and A.2 With one source of AC power to either division inoperable SR 3.8.1.1 (Ref. 2) must be performed within one hour and subsequently every 12 hours after. The inoperable division must be restored to operable status within 30 hours. B.1 With one source of AC power to each division inoperable either division must be restored to two operable sources within 12 hours. C.1 and C.2 If the required diesel generator is inoperable SR 3.8.1.1 must be performed within one hour and subsequently every 12 hours after. Also the required diesel generator must be restored to operable status within 12 hours. D.1 If the required Actions A, B, or C are not met within the completion time then the RCS level must be raised to > EL-117' 0". The will place the plant in a more conservative position with respect to the safety analysis.
Surveillance Requirements	SR 3.10.6.1 Verification of diesel generator operability per SR's 3.8.1.2, 3.8.1.4, 3.8.1.5, 3.8.1.9,

and 3.8.1.18 (Ref. 2) ensures that power will be available during design basis events and for shutdown cooling requirements.

References

1. System 80+ Shutdown Risk Evaluation Report, July 31, 1992, Section 2.4.3.

2. CESSAR-DC, Section 16.11.1.

B 3.10.7 Reduced RCS Inventory Operations - DC Distribution System

Background	The DC distribution systems provide power to the diesel generators. The DC power sources are required in order that the removal of decay heat is not significantly affected.
Applicable Safety Analysis	During reduced RCS inventory operations there are design basis accidents for which DC power is assumed to be available. The DC power sources are required for the mitigation of potential accidents (Ref. 1). The LCO requirements ensure that the DC power sources are available in case of a design basis event.
LCO	One division of the DC distribution system coinciding with the operable diesel generator is required to be operable. It is also required that power is available to the opposite division 125 VDC and 120 VAC distribution centers. These requirements ensure that DC power is available for the mitigation of design basis accidents.
Applicability	LCO 3.10.7 is applicable during MODES 5 and 6 for reduced RCS inventory operations.
Actions	A.1 With a required DC power division inoperable the restoration of the division must be completed within four hours. B.1 With the opposite division distribution centers inoperable, restoration must be completed within 12 hours. C.1 If the required Actions A or B have not been met within the required completion time, then the RCS level must be raised to > EL-117' 0" within 6 hours. This action will place the unit in a more conservative position with respect to the safety analysis.
Surveillance Requirement	SR 3.10.7.1 Performance of SR's 3.8.4.1 through 3.8.4.8 in the frequency specified in Ref. 2 is necessary so that DC power can be made available for the mitigation of design basis accidents.
References	1. CEOG RSTS Criteria Application Vol. 1, Table 2-1. 2. CESSAR-DC, Section 16.11.4

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 Shutdown Margin ~~- T_{avg} > 135°F~~ K

LCO 3.1.1 SHUTDOWN MARGIN (SDM) shall be ≥ [6.5% Δk/k].

-----NOTE-----
With all CEAs verified fully inserted by two diverse position indicators, the CEA of highest reactivity worth does not have to be assumed withdrawn.

APPLICABILITY: MODES 3*, 4*, and 5*. K

* See Special Test Exceptions 3.1.8 and 3.1.10 K

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM ≥ [6.5 % Δk/k].	24 hours

K

Deleted

16.4.2 3.1.2 SHUTDOWN MARGIN - $T_{AVG} \leq 135°F$ K

~~Shutdown Margin - $T_{avg} \leq 135°F$~~ K
3.1.2

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 ~~Shutdown Margin - $T_{avg} \leq 135°F$~~ Deleted

LCO 3.1.2 a. With RTCB's closed, the estimated critical position (ECP) shall be within the limits of technical specifications 3.1.6 and 3.1.7.

b. With RTCB's open, K_{N-1} shall be less than 1.0. K

APPLICABILITY: MODE 5*.

* See special test exceptions 3.1.8 and 3.1.10

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM ≥ [6.5% Δk/k].	24 hours

K

K

SYSTEM 80+ 3.1-2

16.4.~~8~~10 3.1.10 SPECIAL TEST EXCEPTIONS - CEDMS TESTING

3.1 REACTIVITY CONTROL SYSTEMS

3.1.10 Special Test Exception-CEDMS Testing

LCO 3.1.10 The SHUTDOWN MARGIN requirement of Specification 3.1.1 may be suspended for pre-startup tests to demonstrate the OPERABILITY of the control element drive mechanism system (CEDMS) provided:

a. No more than one CEA is withdrawn at any time.

b. No CEA is withdrawn more than [seven] inches.

c. ~~The K_{N-1} requirement of Specification 3.1.2 is met prior to the start of testing.~~ With RTCB's open, K_{N-1} shall be less than ~~1.0~~ [0.99] prior to the start of testing ~~check w/new 3.1 writeup~~

d. All other operations involving positive reactivity changes are suspended during the testing. K

APPLICABILITY: MODES 4 and 5.

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	Any of the above requirements not met.	A.1	Suspend testing and ~~comply with the requirements of 3.1.1 A.~~ Initiate boration to restore SDM to within the limit of LCO 3.1.1	Immediately

Amendment K
October 30, 1992

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.10.1	Determine SDM	Once per 24 hours

* Consider the following factors:

1. RCS boron concentration
2. CEA position
3. RCS average temperature
4. Fuel burnup based on gross thermal energy generation
5. Xenon Concentration
6. Samarium concentration

K

16.6.14 3.3.14 ACCIDENT MONITORING INSTRUMENTATION

3.3 INSTRUMENTATION

3.3.14 Accident Monitoring Instrumentation (AMI)

LCO 3.3.14 The Accident Monitoring Instrumentation specified in Table 3.3.14-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 ~~and~~ 3, and 4*

* APPLICABLE TO INSTRUMENTS ANNOTATED IN TABLE 3.3.14-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Number of channels OPERABLE < Total Number of Channels, but ≥ Minimum Channels Operable requirement of Table 3.3.14-1.	A.1 Restore the inoperable channel to OPERABLE status.	168 hours (7 days)
B. Number of channels OPERABLE less than the Minimum Channels Operable requirement of Table 3.3.14-1.	B.1 Restore one inoperable channel to OPERABLE status.	48 hours
C. Required Actions not met within required Completion Time.	C.1 Be in MODE 4.	12 hours

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December 21, 1990

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.14.1 Perform a CHANNEL CHECK.	31 days
SR 3.3.14.2 Perform a CHANNEL CALIBRATION.	[18 months]

K

TABLE 3.3.14-1

(Sheet 1 of 3)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure (WR)	2	1
(NR)	4	[2]
2. Reactor Coolant Outlet Temperature (T-hot) Wide Range (WR)	4	1
3. Reactor Coolant Inlet Temperature (T-cold) - WR	4	1
4. Reactor Coolant Pressure - WR	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/SG	1/SG
7. Steam Generator (SG) Water Level - WR	2/SG	1/SG
8. In containment Refueling Water Storage Tank Water Level	2	1
9. Emergency Feedwater Flow Rate	2/SG	1
10. Reactor Coolant System Subcooled Margin Monitor	2	1
11. Pressurizer Safety Valve Status	1/valve	1/valve
12. Reactor Vessel Water Level Narrow Range	2	1
13. Core Exit Thermocouples	15/core quadrant	2/core quadrant
14. Emergency Feedwater Storage Tank Water Level	2/tank	1/tank
15. Wide Range Neutron Flux	2	1
16. Reactivity Cavity Level	2	1
17. Containment Area Radiation	2	1
18. Containment Hydrogen Concentration	2	1
19. Containment Isolation Valve Position	1 pair/valve	1 pair/valve

TABLE 3.3.14-1 (Cont'd)

(Sheet 2 of 3)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
20. RCS Radiation Level	2	1
21. Containment Spray Flow	1	1
22. Containment Atmosphere Temperature	2	1
23. Safety Injection Flow	4	2
24. Safety Injection Tank Level	1/tank	1/tank
25. Safety Injection Tank Pressure	1/tank	1/tank
26. Shutdown Cooling Flow	2	2
27. Shutdown Cooling Hx Outlet Temperature	2	2
28. Steam Generator Safety Valve and (ADV) Position	1 pair/valve	1 pair/valve
29. Emergency Ventilation Damper Position	1 pair/damper	1 pair/damper
30. Component Cooling Water Flow to ESF System	1	1
31. Component Cooling Water Temperature to ESF System	1	1
32. DC Bus Voltage	2	2
33. Diesel Generator Voltage	2	2
34. Diesel Generator Current	2	2
35. Diesel Generator Status	2	2
36. 4.16 kV Switchgear Voltage	2	2
37. 480 V Switchgear Voltage	2	2
38. 4.16 kV Switchgear Current	[ ]	[ ]
39. 480 V Switchgear Current	[ ]	[ ]

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TABLE 3.3.14-1 (Cont'd)

(Sheet 3 of 3)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
40. Liquid Blowdown Radiation Monitor (MODES 1, 2, 3, AND 4)	[1/SG]	[1/SG]
41. Main Steam Line Radiation Monitor (MODES 1, 2, 3, AND 4)	[2/SG]	[2/SG]
42. Steam Jet Air Ejector Radiation Monitor (MODES 1, 2, 3, AND 4)	[ ]	1
43. Stack Radiation Monitor (Modes 1, 2, 3, AND 4)	[ ]	1

K

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16.7.3 3.4.3 RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 The combination of RCS pressure, RCS temperature and RCS heatup and cooldown rates shall be maintained within the limits specified in Figure 3.4.3-1.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
---NOTE--- All Required Actions must be completed whenever this Condition is entered.		
A. Requirements of the LCO not met.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Actions and associated Completion Times not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5 with RCS pressure < [500] psig.	6 hours 36 hours

--- NOTE ---
NOT APPLICABLE TO "REGION OF UNALLOWED OPERATION" IN FIGURES 3.4.3-1A AND 3.4.3-1B.

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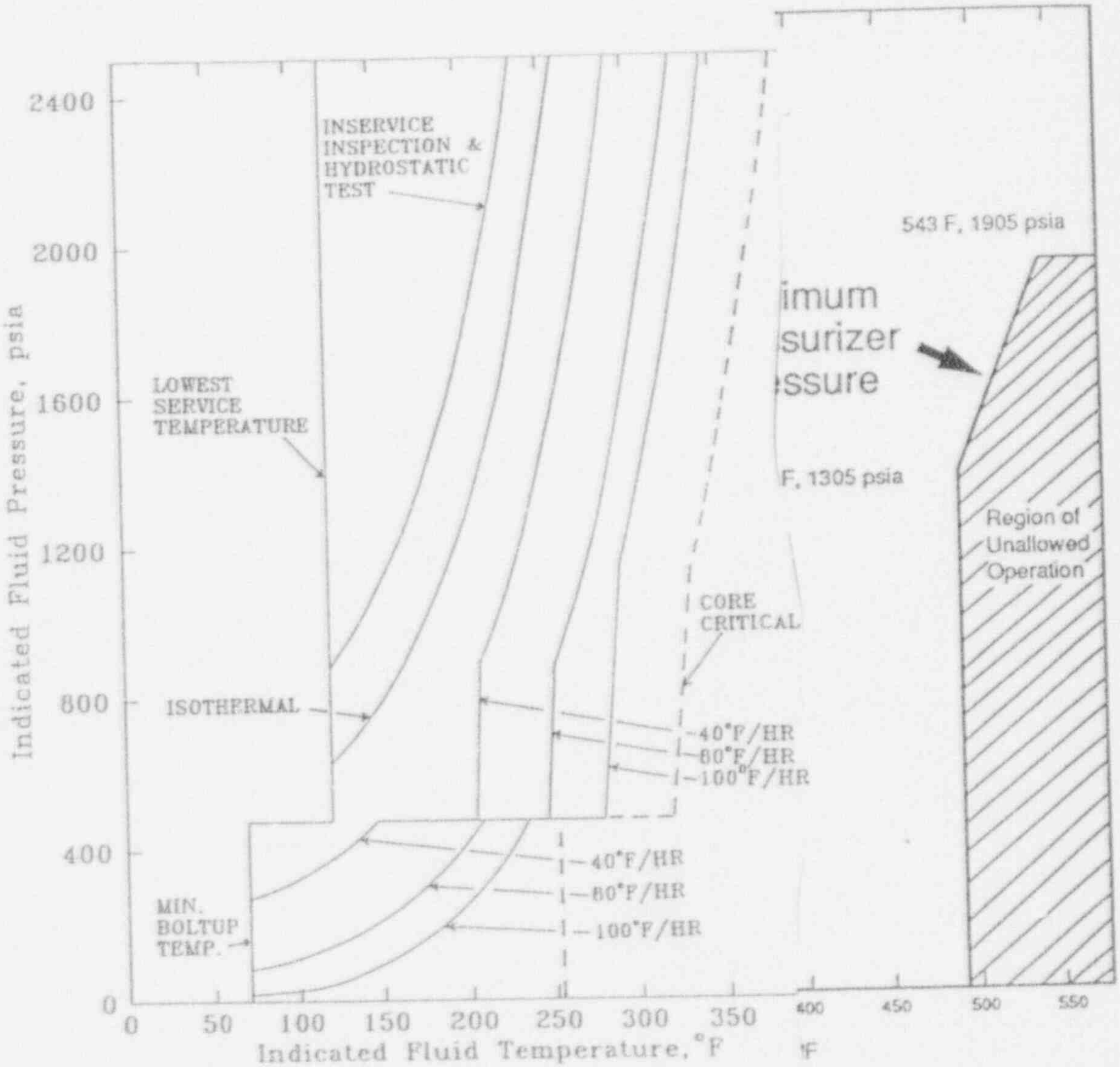
December 21, 1990

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the combination of RCS pressure and temperature and the heatup and cooldown rates within limits.	--------NOTE-------- - Only required during RCS heatup and cooldown operations and inservice leak and hydrostatic testing. --------------------- 30 minutes

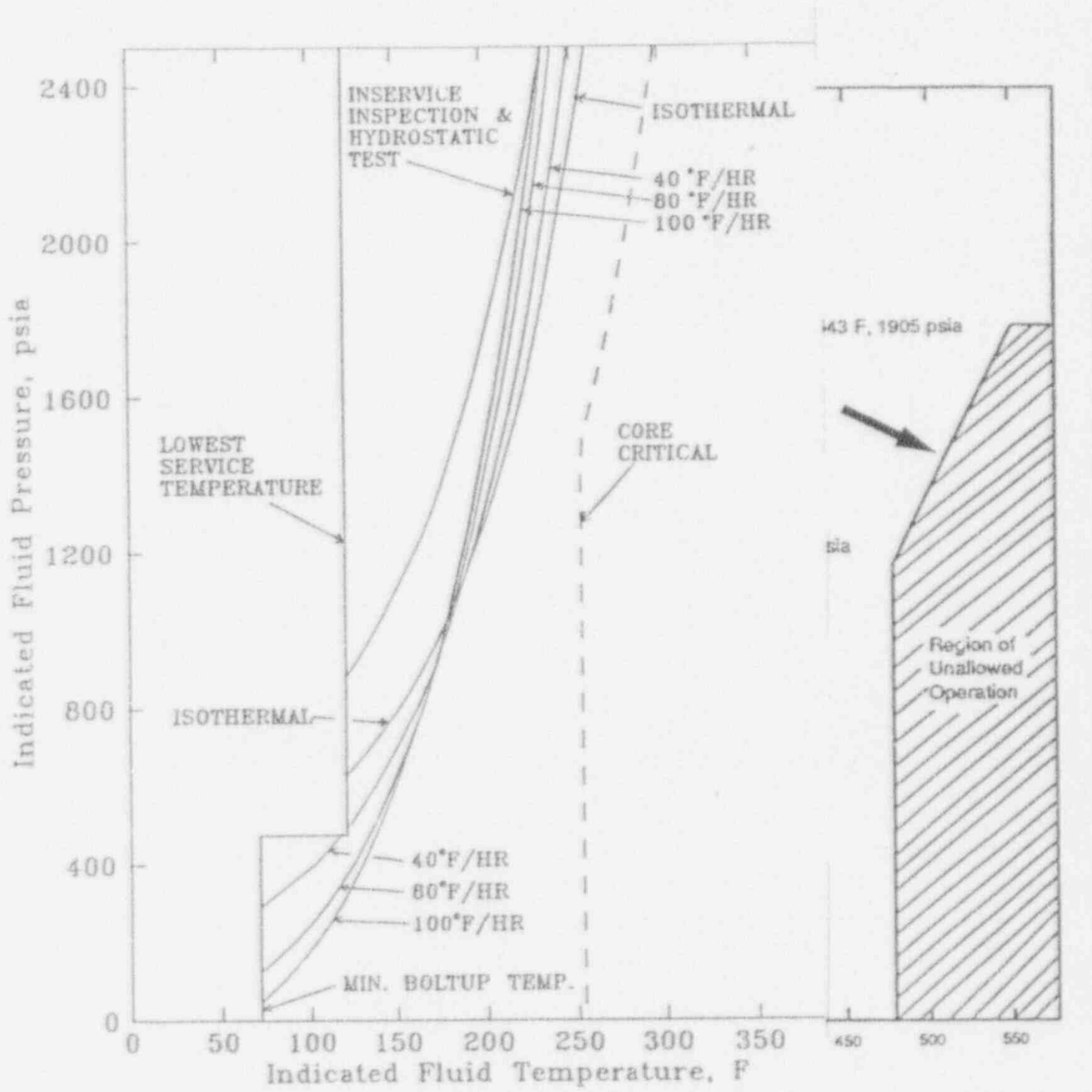
K



RCS PRESSURE AND TEMPERATURE LIMITS
(HEAT-UP)

FIGURE 3.4.3-1A

K



RCS PRESSURE AND TEMPERATURE LIMITS
(COOLDOWN)

FIGURE 3.4.3-1B

K

16.7.6 3.4.6 RCS LOOPS - MODE 4

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

LCO 3.4.6 Two RCS loops/Shutdown Cooling System (SCS) divisions consisting of any combination of RCS loops and SCS divisions shall be OPERABLE and at least one loop/division shall be in operation. K

---NOTE---

1. All RCPs and SCS pumps may be de-energized for up to 1 hour per 8-hour period provided: K

   a. No operations are permitted that would cause reduction of the RCS boron concentration, and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. No RCP shall be started with any RCS cold leg temperatures ≤ [317°F] ~~unless~~ [259°F] during cooldown or [290°F] during heatup (the heatup rate is limited to [40°F/hr or less]) unless The secondary water temperature of each steam generator is < [100°F] above each of the RCS cold leg temperatures.

APPLICABILITY: MODE 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCS loop inoperable AND Two SCS divisions inoperable.	A.1 Initiate action to return a second RCS loop/SCS division to OPERABLE status.	Immediately

K
K

(continued)

16.7.7 3.4.7 REACTOR COOLANT LOOPS AND CIRCULATION - MODE 5, LOOPS FILLED

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 <u>Reactor Coolant Loops - MODE 5, Loops Filled</u> K

LCO 3.4.7 One Shutdown Cooling System (SCS) division shall be OPERABLE and in operation, and either: K

a. One additional SCS division shall be OPERABLE, or K

b. The secondary side water level of each Steam Generator (SG) shall be ≥ [25%] wide range indication. K

-----------------------------------NOTE-----------------------------------

1. SCS pumps may be de-energized for up to 1 hour per 8-hour period provided: K

   a. No operations are permitted that would cause reduction of the RCS boron concentration, and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. No RCP shall be started with one or more of the RCS cold leg temperatures ≤ ~~[317°F]~~ unless:

   a. The secondary water temperature of each SG is < [100°F] above each of the RCS cold leg temperatures.

3. All SCS trains may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation. K

APPLICABILITY: MODE 5 with RCS loops filled.

[259°F] during cooldown or [290°F] during heatup (the heatup rate is limited to [40°F/hr or less]

16A.7.8 B 3.4.8 RCS LOOPS - MODE 5 (LOOPS NOT FILLED)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5 (Loops Not Filled)

BASES

BACKGROUND

In MODE 5 with the Reactor Coolant System (RCS) loops not filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the shutdown cooling ~~(SDC)~~ SCS heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only the ~~SDC system~~ SCS can be used for coolant circulation. The number of divisions in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one ~~SDC~~ SCS division for decay heat removal and transport. The other intent of this LCO is to require that two paths be available to provide redundancy for heat removal.

This LCO permits limited periods without forced circulation. When the ~~SDC~~ SCS divisions are not in operation, no alternate heat removal path exists. The response of the RCS without the ~~SDC system~~ SCS depends on the decay heat load and the length of time that the ~~SDC~~ SCS pumps are stopped. As decay heat diminishes, the effects on RCS temperature diminish. Without cooling by ~~SDC,~~ SCS higher heat loads will cause the reactor coolant temperature to increase at a rate proportional to the decay heat load. Because pressure can increase, applicable system pressure limits (pressure and temperature limits or low temperature overpressurization limits) must be observed and forced ~~SDC system~~ SCS flow must be reestablished prior to reaching the pressure limit. Entry into a condition with no ~~SDC~~ SCS division in operation stops heat removal and should only be considered for limited circumstances such as when switching from one ~~SDC~~ SCS division to the other. With the pumps stopped, pressure and temperature may increase and pumps must be restored prior to exceeding pressure and subcooling limits.

(continued)

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December 21, 1990

16.7.10 3.4.10 PRESSURIZER SAFETY VALVES

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Four pressurizer safety valves shall be OPERABLE with lift settings ≥ [2475 psia] and ≤ [2525 psia].

-----NOTE-----
LCO 3.0.4 and SR 3.0.4 are not applicable for entry into MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for [72] hours following entry into MODE 3, provided a preliminary cold setting was made prior to heatup.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4 with any RCS cold leg temperature > ~~[317°F]~~.
[259°F] during cooldown or [390°F] during heatup (the heatup rate limited to [40°F/hr or less])

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One pressurizer code safety valve inoperable.	A.1	Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 4 with all RCS cold leg temperatures ≤ [317°F] [259°F]	[12 hours]

OR

B.3 Be in MODE 4 on Shutdown Cooling with the requirements of LCO 3.4.11 met.

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16.7.11 3.4.11 LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.11 LTOP System shall be OPERABLE as follows: K

a. Two SCS Relief Valves with lift settings ≤ [530] ~~[550]~~ psig and associated block valves open, or

b. The RCS depressurized with both divisions of Rapid Depressurization (RD) valves open.

APPLICABILITY: MODE 4, with any RCS cold leg temperature ≤ [317°F],
MODE 5,
MODE 6, with the reactor vessel head on

-----NOTE-----
LCO 3.0.4 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SCS Relief Valve inoperable.	A.1 Restore SCS Relief Valve to OPERABLE status.	24 hours*

(continued)

*MODE 4 completion time 7 days. K

MODE 4, with any RCS cold leg temperature less than or equal to:
a. [259°F] during cooldown or
b. [290°F] during heatup
(The heatup rate is limited to [40°F/hr or less])

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Actions and associated Completion Times not met. <u>OR</u> Both SCS Relief Valves inoperable.	B.1 Depressurize RCS and open Rapid Depressurization Valves.	8 hours
C. Both SCS Relief Valves inoperable. <u>AND</u> One or more Rapid Depressurization Valves closed.	C.1.1 Initiate action to OPEN closed Rapid Depressurization valve. <u>OR</u> C.1.2 Establish a vent path of $\geq$ [1.3 in^2].	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	-----NOTE----- Only required when complying with Required Action B.1. --------------- Verify required RD Valves open.	12 hours
SR 3.4.11.2	Verify that the block valve is open for each required SCS Relief Valve.	12 hours
SR 3.4.11.3	Perform a SETPOINT CALIBRATION for each required SCS Relief Valve.	[18 months]

?

?

?

K

K

K

K

16.7.17 3.4.17 REACTOR COOLANT GAS VENT SYSTEM

3.4 REACTOR COOLANT SYSTEM

3.4.17 Reactor Coolant Gas Vent System

LCO 3.4.17 Both reactor coolant system gas vent paths shall be OPERABLE and closed at each of the following locations:

a. Reactor vessel head, and

b. Pressurizer steam space

APPLICABILITY: MODE 1, 2, 3 and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Vent path inoperable from the reactor head vent line.	A.1 Restore required vent path to OPERABLE status.	72 hours
	OR	
	A.2 Be in MODE 3.	6 hours
	AND	
	A.3 Be in MODE 5.	36 hours
B. Vent path inoperable from the pressurizer steam space vent line.	B.1 Restore required vent path to OPERABLE status.	72 hours
	OR	
	B.2 Be in Mode 3.	6 hours
	AND	
	B.3 Be in Mode 5.	36 hours

(continued)

K

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. None of the required reactor coolant system vent paths OPERABLE.	C.1 Restore at least one of the required vent paths to OPERABLE status.	6 hours
	<u>OR</u>	
	C.2 Be in Mode 3.	6 hours
	<u>AND</u>	
	C.3 Be in Mode 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify all manual isolation valves in each vent path are locked in the open position.	18 months
SR 3.4.17.2	Cycle each vent through at least one complete cycle from the control room.	18 months
SR 3.4.17.3 4	Verify flow through the reactor coolant system vent paths during venting.	18 months
SR 3.4.17.4 5	Verify correct breaker alignment and position indication power available.	7 days
SR 3.4.17.3	Verify all manual isolation valves to pressure instruments in each vent path are in the open position	18 months

K

16.7.18 3.4.18 RAPID DEPRESSURIZATION FUNCTION

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.18 Rapid Depressurization Function

LCO 3.4.18 At least one of the two vent paths providing the Rapid Depressurization Function of the Safety Depressurization System shall be operable and both paths shall be closed at the Pressurizer steam space.

APPLICABILITY: MODE 1, 2, 3, & 4 AND ~~5~~ 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Two or more valves inoperable causing both vent paths to be inoperable	A.1 Restore at least one of the two vent paths to OPERABLE status. OR A.2 Be in MODE3 OR A.3 Be in MODE 5	~~6~~ 72 hours 6 hours ~~6~~ 36 hours

<INSERT B>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.18.1	Verify all manual isolation valves to pressure instruments in each vent path are in the open position.	18 months
SR 3.4.18.2	Cycle the valves in each vent path at least one complete cycle from the control room.	18 months
SR 3.4.18.3	Verify correct valve position indication in the control room for all valves.	7 days
SR 3.4.18.4	Verify correct breaker alignment and position indication power indication power available. ~~Note: Breakers are normally ON for the gate valves and OFF for the globe valves.~~	7 days

16.9.8 3.6.8 REACTOR SHIELD BUILDING K

3.6 CONTAINMENT SYSTEMS

3.6.8 Reactor Shield Building K

LCO 3.6.8 Reactor Shield Building shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor Shield Building inoperable.	A.1 Restore reactor shielding building to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

K

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.1	Verify that each door in each access opening is closed except when the access opening is being used for normal ~~transient~~ transit entry and exit, then at least one door shall be closed.	31 days
SR 3.6.8.2	Verify Reactor Shield Building structural integrity by performing a visual inspection of the exposed interior and exterior surfaces of the Reactor Shield Building.	During shutdown for SR 3.6.1.1 Type A tests

K
K

16.10.18 3.7.18 PLANT SYSTEMS

3.7 PLANT SYSTEMS

3.7.18 Nuclear Annex Ventilation System

LCO 3.7.18 All Essential Mechanical Equipment Room Cooling and Ventilation Units in the Nuclear Annex shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, ~~and~~ 4, 5 and 6

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One Essential Mechanical Equipment Room Cooling and Ventilation unit inoperable.	A.1	Restore inoperable unit to OPERABLE status.	7 days
B.	Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
		AND		
		B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	Operate each Essential Mechanical Equipment Room Cooling and Ventilation unit for ≥ 15 minutes.	31 days during periods when the system has not been operated.
SR 3.7.18.2	Demonstrate one Nuclear Annex Building Essential Mechanical Equipment Room Cooling and Ventilation unit actuates on an actual or simulated actuation signal.	[18 months]

K

16.13.2 3.10.2 REDUCED RCS INVENTORY OPERATIONS - INSTRUMENTATION

3.10 REDUCED RCS INVENTORY OPERATIONS

3.10.2 Reduced RCS Inventory Operations - Instrumentation

LCO 3.10.2 The following reactor coolant system instrumentation shall be operable.

a. Two independent means of monitoring RCS level indications; one narrow range and one wide range instrument. And,

b. Two independent ~~and diverse~~ means of monitoring RCS temperature. And,

c. Two independent indications available to monitor SCS performance in the loop on service for decay heat removal.

APPLICABILITY: MODE 5 REDUCED RCS INVENTORY

AND

MODE 6 REDUCED RCS INVENTORY K

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. WR RCS Level Indication inoperable	A.1 Initiate action to restore Instrument to OPERABLE Status. And,	[Immediately]
	A. Monitor RCS Temp	[Every 30 minutes]
	B. Monitor SCS Performance	[Every 30 minutes]
	C. Monitor NR Level	[Every 30 minutes]

ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. NR RCS Level Indication inoperable	B.1 Initiate action to restore Instrument to OPERABLE Status. And,	[Immediately]
	A. Monitor RCS Temp B. Monitor SCS Performance C. Monitor WR Level	[Every 30 minutes] [Every 30 minutes] [Every 10 minutes]
	<u>AND</u>	[Immediately]
	B.2 Initiate action to restore RCS level to > [EL-117'0"]	
C. One RCS Temperature Indication inoperable	C.1 Initiate action to restore instrument indication to OPERABLE status. And,	[Immediately]
	A. Monitor RCS Level B. Monitor SCS Performance C. Monitor OPERABLE Temperature Instrument	[Every 30 minutes] [Every 30 minutes] [Every 30 minutes]
D. Two RCS Temperature Indication inoperable	D.1 Initiate Action to restore one instrument to OPERABLE status. And,	[6 hours] [Immediately]
	A. Monitor RCS level B. Monitor SCS performance	[Every 10 minutes] [Every 10 minutes]
	<u>AND</u>	[Immediately]
	D.2 Initiate action to restore RCS level to >[EL-117'0"]	

K

E. SCS Performance Indications inoperable	E.1 Initiate action to restore instrument to OPERABLE status.	[Immediately]
	AND →	
	A. Monitor RCS Temp	[Every 10 minutes]
	B. Monitor RCS Level	[Every 10 minutes]
	OR	
	E.2 Align decay heat removal systems to the alternate loop.	[2 hours]

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.2.1	Perform a CHANNEL CHECK of RCS Level; One WR and One NR.	[6 hours]
SR 3.10.2.2	Perform a CHANNEL CHECK of RCS Temperature	[6 hours]
SR 3.10.2.3	Perform a CHANNEL CHECK of SCS performance in the loop removing decay heat.	[6 hours] K
SR 3.10.2.4	Perform a CHANNEL CALIBRATION of the applicable RCS level, temperature and SCS performance.	[60 days]

16.13.4 3.10.4 REDUCED RCS INVENTORY OPERATIONS - HEAT REMOVAL

3.10 REDUCED RCS INVENTORY OPERATIONS

3.10.4 Reduced RCS Inventory Operations - Heat Removal

LCO 3.10.4 K

a. Two Shutdown Cooling System (SCS) divisions shall be OPERABLE, and at least one division shall be in operation.

b. The Containment Spray pump shall be OPERABLE in the operating division.

APPLICABILITY: MODE 5 REDUCED RCS INVENTORY

AND

MODE 6 REDUCED RCS INVENTORY

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SCS division inoperable.	A.1 Restore division to OPERABLE status.	[~~Immediately~~] 15 minutes
B. Required Action and associated Completion Time not met.	B.1 Raise RCS level to > [EL-117'0"].	[6 hours]

ACTIONS (Continued)

CONDITION		REQUIRED ACTION	COMPLETION TIME
C.	No SCS division in operation.	C.1 Suspend all operations involving reduction in RCS boron concentration.	[Immediately]
		AND	
		C.2 Initiate action to restore one SCS division to OPERABLE status and place in operation.	[Immediately]
		AND	
		C.3 Initiate action to raise RCS level to >[EL-117'0"]	[Immediately]
D.	Containment Spray Pump in operating division inoperable	D.1 Initiate action to place the alternate division in operation if the containment spray pump in that division is OPERABLE.	[6 hours]
		AND	
		D.2 Monitor SCS performance.	[Every 30 minutes]
		AND	
		D.3 Restore inoperable Containment Spray Pump.	[48 hours]
E.	Required Action and Completion time of Item D.3 not met.	E.1 Raise RCS Level >[EL-117'0"]	[6 hours]

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.4.1	Verify at least one SCS division operating	[12 hours]
SR 3.10.4.2	Verify correct breaker alignment and indicated power available to the SCS pump that is not in operation and the OPERABLE Containment Spray pump.	[24 hours]

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