ATTACHMENT 3

LICENSE AMENDMENT REQUEST 244:
PROPOSED REVISION TO RADIOLOGICAL CONSEQUENCES ANALYSIS AND CONTROL ROOM HABITABILITY TECHNICAL SPECIFICATIONS

MARKED-UP TECHNICAL SPECIFICATIONS BASES PAGES

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

B 3.3 INSTRUMENTATION

B 3.3.6 Containment Purge and Vent Isolation Instrumentation

BASES

BACKGROUND

Containment purge and vent isolation instrumentation closes the containment isolation valves in the Containment Vessel Air Handling System, consisting of the Containment Air Cooling and Containment Purge and Vent Systems. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Containment Air Cooling System may be in use during reactor operation and the Containment Purge and Vent System will be in use with the reactor shutdown.

Containment purge and vent isolation initiates on an automatic safety injection (SI) signal; a manual SI signal; a manual containment vent isolation signal; or a manual containment spray signal (of both trains). The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss these modes of initiation.

Three radiation monitoring channels are also provided as input to the containment purge and vent isolation. The three channels measure containment radiation at two locations. One channel is a particulate monitor (R-11), the second channel is a radioactive gas monitor (R-12), and the third channel is also a radioactive gas monitor (R-21). The three channels are separated into two trains with channel R-21 designated as Train A and channels R-11 and R-12 designated as Train B. All three detectors will respond to most events that release radiation to containment. However, analyses have not been conducted to demonstrate that all credible events will be detected by more than one monitor. Therefore, for the purpose of this LCO the three channels are not considered redundant. Since the radiation monitors constitute a sampling system, various components such as sample line valves, sample line heaters, sample pumps, and filter motors are required to support monitor OPERABILITY.

Each of the purge systems has inner and outer containment isolation valves in its supply and exhaust ducts. A high radiation signal from any one of the three channels initiates containment purge isolation, which closes both inner and outer containment isolation valves in the Containment Purge and Vent System and the 2 inch containment vent isolation valves. These valves are described in the Bases for LCO 3.6.3, "Containment Isolation Valves."

BASES

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event. The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment purge and vent isolation radiation monitors act as backup to the SI signal to ensure closing of the purge and vent valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown. Containment isolation contributes to both meeting the containment leakage rate assumptions of the safety analyses and ensuring that the calculated control room and accidental offsite radiological doses are below 10 CFR 50.67 (Ref. 1) limits. Due to radioactive decay, containment is only required to isolate during fuel handling accidents involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours).

The containment purge and vent isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Purge and Vent Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Automatic Actuation Logic and Actuation Relays

   The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

   Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI, and ESFAS Function 3, Containment Isolation. The applicable MODES and specified conditions for the containment purge and vent isolation portion of these Functions are different than those for their Containment isolation and SI roles. If one or more of the SI or Containment isolation Functions becomes inoperable in such a manner that only the Containment Purge and Vent Isolation Function is affected, the Conditions applicable to their SI and Containment isolation Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Purge and Vent Isolation Functions specify sufficient compensatory measures for this case.

2. Containment Radiation

   The LCO specifies three required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment Purge and Vent Isolation remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY will also

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LCO (continued)

require correct valve lineups, sample pump operation, and filter motor operation, as well as detector OPERABILITY, since these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

The radioactive gas monitor (R-21) has two flow path alignments; it can be aligned to the 36 inch containment purge exhaust line or to the containment atmosphere via the same penetration used by particulate monitor R-11 and radioactive gas monitor R-12. However, since the 36 inch containment purge exhaust line is isolated and sealed in MODES 1, 2, 3, and 4, for the radioactive gas monitor R-21 to be OPERABLE, it must be aligned to the containment atmosphere via the same containment penetration as the R-11 and R-12 radiation monitors.

3. Containment Isolation - Manual Initiation

Refer to LCO 3.3.2, Function 3.a, for all initiating Functions and requirements. This Function provides the manual initiation capability for containment ventilation isolation.

4. Containment Spray - Manual Initiation

Refer to LCO 3.3.2, Function 2.a, for all initiating Functions and requirements. This Function provides the manual initiation capability for containment ventilation isolation.

5. Safety Injection

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements. This Function provides both manual and automatic initiation capability for containment ventilation isolation.

APPLICABILITY

The Automatic Actuation Logic and Actuation Relays and Containment Radiation Functions are required OPERABLE in MODES 1, 2, 3, and 4, and during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) within containment. Under these conditions, the potential exists for a release of fission product radioactivity into containment. Therefore, the containment purge and vent isolation instrumentation must be OPERABLE in these MODES.

While in MODES 5 and 6, the containment purge and vent isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to

BASES

APPLICABILITY (continued)

ensure post accident offsite doses are maintained within the limits of Reference 1.

The Applicability for the containment purge and vent isolation on the Containment Isolation - Manual Initiation, Containment Spray - Manual Initiation, and Safety Injection Functions are specified in LCO 3.3.2. Refer to the Bases for LCO 3.3.2 for discussion of the Containment Isolation - Manual Initiation, Containment Spray - Manual Initiation, and Safety Injection Function Applicability.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one containment radiation monitor channel. Since the three containment radiation monitors measure different parameters, failure of a single channel may result in loss of the radiation monitoring Function for certain events. Consequently, the failed channel must be restored to OPERABLE status. The 4 hours allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that one or more of the remaining channels will respond to most events.

B.1

Condition B applies to one or more Automatic Actuation Logic and Actuation Relays trains and addresses the train orientation of the master and slave relays for these Functions. It also addresses the failure of

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ACTIONS

B.1 (continued)

multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4.

C.1 and C.2

Condition C applies to one or more Automatic Actuation Logic and Actuation Relays trains and addresses the train orientation of the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1. If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action to place and maintain containment purge and vent isolation valves in their closed position is met or the applicable Conditions of LCO 3.9.6, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.

A Note states that Condition C is applicable during movement of recently irradiated fuel assemblies within containment.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Purge and Vent Isolation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that

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SURVEILLANCE REQUIREMENTS

SR 3.3.6.1 (continued)

instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.6.2

SR 3.3.6.2 is the performance of an ACTUATION LOGIC TEST. The train being tested is placed in the test condition, thus preventing inadvertent actuation. All possible logic combinations are tested for each protection function. This test is performed every 92 days on a STAGGERED TEST BASIS. The Surveillance interval is justified in Reference 2.

The SR is modified by a Note stating that the Surveillance is only applicable to the actuation logic of the ESFAS Instrumentation.

SR 3.3.6.3

A COT is performed every 92 days on each required channel to ensure the entire channel will perform the intended Function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable

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SURVEILLANCE REQUIREMENTS

SR 3.3.6.3 (continued)

because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 3). This test verifies the capability of the instrumentation to provide the containment purge and vent system isolation. The Setpoint Control Program (SCP) has controls which require verification that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

SR 3.3.6.4

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The SCP has controls which require verification that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 50.67.

2. WCAP-15376, Rev. 0, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," October 2000.

3. NUREG-1366, December 1992.

B 3.3 INSTRUMENTATION

B 3.3.7 Control Room Post Accident Recirculation (CRPAR) System Actuation Instrumentation

BASES

BACKGROUND

The CRPAR System provides an enclosed control room environment from which the unit can be operated following an uncontrolled release of radioactivity. The CRPAR System is part of the Control Room Air Conditioning System. During normal unit operation, the Control Room Air Conditioning System provides cooling and heating of recirculated and fresh air to ventilate the control room. Upon receipt of an actuation signal, both CRPAR fans are started, the flow path through the Emergency Filtration System is opened, and a portion of the return air volume is filtered to remove airborne contaminants and airborne radioactivity, then mixed with the recirculated return air. This system is described in the Bases for LCO 3.7.10, "Control Room Post Accident Recirculation (CRPAR) System."

~~The actuation instrumentation consists of a single radiation monitor (R-23) located on the common discharge of the outlet of the air conditioning fan units. A high radiation signal from the detector will initiate both trains of the CRPAR System. The control room operator can also start the CRPAR fan(s) by manual switches in the control room.~~ The CRPAR System is ~~also~~ actuated by a safety injection (SI) signal. The control room operator can also start the CRPAR fan(s) by manual switches in the control room. The SI Function is discussed in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

APPLICABLE SAFETY ANALYSES

The control room must be kept habitable for the operators stationed there during accident recovery and post accident operations.

The CRPAR System acts to terminate the normal supply of unfiltered outside air to the control room, both CRPAR fans are started, the flow path through the Emergency Filtration System is opened, and a portion of the return air volume is filtered to remove airborne contaminants and airborne radioactivity, then mixed with the recirculated return air. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post accident operations by minimizing the radiation exposure of control room personnel.

~~The radiation monitor~~Manual actuation of the CRPAR System is a backup for the SI signal actuation. This ensures initiation of the CRPAR System during a loss of coolant accident or steam generator tube rupture when an initiation of SI is anticipated. In addition, ~~the radiation monitor~~ manual actuation of the CRPAR System is the primary means to ensure control room habitability in the event of a locked rotor accident.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~The radiation monitor~~Manual actuation of the CRPAR System ~~in MODES 5 and 6,~~ and a requirement for the control room envelope to be isolated during movement of recently irradiated fuel assemblies (TS 3.7.10) is the primary means to ensure control room habitability in the event of a fuel handling~~, volume control tank, or waste gas decay tank rupture~~ accident.

The CRPAR System actuation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that instrumentation necessary to initiate the CRPAR System is OPERABLE.

1. Automatic Actuation Logic and Actuation Relays

   The LCO requires two trains of Actuation Logic and Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

   Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b., SI, in LCO 3.3.2 and include the slave relays that send the SI signal to the CRPAR System. The applicable MODES and specified conditions for the CRPAR System portion of these functions are different than those specified for their SI roles. If one or more of the SI functions becomes inoperable in such a manner that only the CRPAR System function is affected, the Conditions applicable to their SI function need not be entered. The less restrictive Actions specified for inoperability of the CRPAR System Functions specify sufficient compensatory measures for this case.

~~2. Control Room Vent Radiation Monitor~~

   ~~The LCO specifies one Control Room Vent Radiation Monitor to ensure that the radiation monitoring instrumentation necessary to initiate the CRPAR System remains OPERABLE.~~

   ~~For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, and filter motor operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.~~

3. Safety Injection

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements.

BASES

APPLICABILITY

The CRPAR Functions must be OPERABLE in MODES 1, 2, 3, 4, and during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of the critical reactor core within the previous 375 hours). ~~The Functions must also be OPERABLE in MODES 5 and 6 when required for a waste gas decay tank rupture accident, to ensure a habitable environment for the control room operators.~~

The Applicability for the CRPAR actuation on the ESFAS Safety Injection Functions are specified in LCO 3.3.2. Refer to the Bases for LCO 3.3.2 for discussion of the Safety Injection Function Applicability.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A.1

Condition A applies to the Automatic Actuation Logic and Actuation Relays Function of the CRPAR System.

If one train is inoperable, 7 days are permitted to restore it to OPERABLE status. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this Completion Time is the same as provided in LCO 3.7.10. If the train cannot be restored to OPERABLE status, the associated CRPAR train must be placed in the emergency mode of operation. This accomplishes the actuation instrumentation Function and places the unit in a conservative mode of operation.

B.1.1, B.1.2, and B.2

Condition B applies to the failure of two Automatic Actuation Logic and Actuation Relay trains ~~or the Control Room Vent Radiation Monitor~~. The first Required Action is to place one CRPAR train in the emergency mode of operation immediately. This accomplishes the actuation instrumentation Function that may have been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required

BASES

ACTIONS

B.1.1, B.1.2, and B.2 (continued)

Actions of LCO 3.7.10 must also be entered for the CRPAR train made inoperable by the inoperable actuation instrumentation and not placed in the emergency mode of operation. This ensures appropriate limits are placed upon train inoperability as discussed in the Bases for LCO 3.7.10.

Alternatively, both CRPAR trains may be placed in the emergency mode. This ensures the CRPAR function is performed even in the presence of a single failure.

C.1 and C.2

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and the unit is in MODE 1, 2, 3, or 4. The unit must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1

Condition D applies when the Required Action and associated Completion Time for Condition A or B have not been met when recently irradiated fuel assemblies are being moved. Movement of recently irradiated fuel assemblies must be suspended immediately to reduce the risk of accidents that would require CRPAR System actuation.

~~E.1~~

~~Condition E applies when the Required Action and associated Completion Time for Condition A or B have not been met in MODE 5 or 6. Actions must be initiated to restore the inoperable train(s) to OPERABLE status immediately to ensure adequate isolation capability in the event of a waste gas decay tank rupture.~~

BASES

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.7-1 determines which SRs apply to which CRPAR System Actuation Functions.

~~SR 3.3.7.1~~

~~Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.~~

~~Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.~~

~~The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.~~

~~SR 3.3.7.2~~

~~A COT is performed once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the CRPAR System actuation. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Setpoint Control Program (SCP) has controls which require verification that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. The Frequency is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.~~3~~1

SR 3.3.7.3 is the performance of an ACTUATION LOGIC TEST. For the portion of the logic common to ESFAS, Function 1.b ACTUATION LOGIC TEST, the train being tested is placed in the test condition, thus preventing inadvertent actuation and all possible SI logic combinations are tested for each protection function. For the portion of the logic not tested as part of the ESFAS Function 1.b ACTUATION LOGIC TEST (i.e., the slave relay), actuation of the end devices may occur. The Frequency of 18 months is based on the refueling outage cycle, since the slave relay cannot be tested at power without resulting in actuation of affected components.

The SR is modified by a Note stating that the Surveillance is only applicable to the actuation logic of the ESFAS Instrumentation.

~~SR 3.3.7.4~~

~~A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.~~

~~The SCP has controls which require verification that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.~~

~~The Frequency is based on operating experience and is consistent with the typical industry refueling cycle~~.

REFERENCES

1. WCAP-15376, Rev. 0, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," October 2000.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

USAR General Design Criteria (GDC) 16 (Ref. 1) states that means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. USAR, Section 6.5 (Ref. 2) describes the capabilities of the leakage monitoring indication systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from the steam generators (SGs) is 150 gallons per day per SG. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is a condition assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. Other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR), locked reactor coolant pump rotor, and control rod ejection. The primary to secondary leakage contaminates the secondary fluid.

The radiological accident analysis ~~(Ref. 3)~~ for SGTR assumes the contaminated secondary fluid is released to the environment from the ruptured and the intact SGs. The release from the ruptured SG occurs until ~~30~~ 55 minutes after the reactor trip and the release from the intact SG occurs until ~~24~~ 29 hours after the reactor trip when residual heat removal is placed in service. The 150 gallons per day SG primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential.

~~The SLB is less limiting for site radiation releases.~~ The safety analysis for the SLB accident assumes the 150 gallons per day primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB accident are ~~well~~ within the limits defined in 10 CFR 50.67 or the staff approved licensing basis (i.e., ~~a small fraction of these~~RG 1.183, Rev 0 limits).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. 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. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

BASES

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

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 required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

BASES

APPLICABILITY (continued)

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limit, or if unidentified or identified LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures 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. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.2 (continued)

operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

REFERENCES

1. USAR, Section 4.1.3.2, GDC 16, "Monitoring Reactor Coolant Leakage."

2. USAR, Section 6.5, Leakage Detection and Provisions for the Primary and Auxiliary Coolant Loops.

3. ~~Westinghouse calculation CN-CRA-99-36, Steam Generator Tube Rupture.~~ Not Used

4. NEI 97-06, "Steam Generator Program Guidelines."

5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR 50.67 (Ref. 1). Doses to control room operators must be limited per 10 CFR 50, GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a main steam line break (MSLB) or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in Regulatory Guide (RG) 1.183 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate RG 1.183 acceptance criteria following a MSLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits and an existing reactor coolant steam generator (SG) tube leakage rate of 150 gallons per day exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of ~~0.1~~ 0.05 μCi/gm DOSE EQUIVALENT I-131 from LCO ~~3.7.17~~ 3.7.16, "Secondary Specific Activity."

The analyses for the MSLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at ~~1.0~~ 0.1 μCi/gm DOSE EQUIVALENT I-131 with an accident initiated iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a MSLB ~~or SGTR~~ (by a factor of 500) or a SGTR (by a factor of 335). The second case assumes the initial reactor coolant iodine activity at ~~20.0~~ 10.0 μCi/gm DOSE EQUIVALENT I-131 for the MSLB accident and ~~20.0~~ 10.0 μCi/gm DOSE EQUIVALENT I-131 for the SGTR accident due

BASES

APPLICABLE SAFETY ANALYSES (continued)

to an iodine spike caused by a reactor or an RCS transient prior to the accident. In both cases, the noble gas specific activity is assumed to be ~~595~~ 16.4 μCi/gm DOSE EQUIVALENT XE-133.

The SGTR analysis also considers a possible loss of offsite power at the same time as the reactor trip. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

A coincident loss of offsite power would cause the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG is assumed to discharge radioactively contaminated steam to the atmosphere through the ~~main steam safety valves~~ SG power operated relief valves. The unaffected SG removes core decay heat by venting steam to the atmosphere until the event is terminated.

The MSLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. Reactor trip occurs after the generation of an SI signal on low steam line pressure. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SG removes core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR System is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed ~~20.0~~ 10.0 μCi/gm for more than 48 hours.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The iodine specific activity in the reactor coolant is limited to ~~1.0~~ 0.1 μCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to ~~595~~ 16.4 μCi/gm DOSE EQUIVALENT XE-133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate RG 1.183 acceptance criteria (Ref. 2)

The MSLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a MSLB or SGTR, lead to doses that exceed the RG 1.183 acceptance criteria (Ref. 2).

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a MSLB or SGTR to within the RG 1.183 acceptance criteria (Ref. 2).

In MODES 5 and 6, the steam generators are not normally being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is ≤ ~~20~~ 10 μCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limit within 48 hours. The Completion Time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a MSLB or SGTR occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) relying on Required Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

B.1

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limit within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a MSLB or SGTR occurring during this time period.

BASES

ACTIONS

B.1 (continued)

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Action B.1 while the DOSE EQUIVALENT XE-133 LCO limit is met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B is not met, or if the DOSE EQUIVALENT I-131 is > ~~20~~ 10 μCi/gm, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in noble gas specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The 7 day Frequency considers the low probability of a gross fuel failure during the time.

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.16.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.2

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during operation and following fast power changes when iodine spiking is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

REFERENCES

1. 10 CFR 50.67.

2. Regulatory Guide 1.183, July 2000.

3. USAR, Section 14.2.4.

4. USAR, Section 14.2.5.

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Post Accident Recirculation (CRPAR) System

BASES

BACKGROUND

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

The CRPAR System consists of two independent, redundant trains that recirculate and filter the air in the control room envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air~~outside air~~. Each CRPAR train consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Common ductwork, valves or dampers, doors, barriers, and instrumentation also form part of the system.

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

The CRPAR System is an emergency system, which is normally in the standby mode of operation. The CRPAR System is part of the Control Room Air Conditioning (CRAC) System. During normal unit operation, the CRAC System provides cooling of recirculated and fresh air to ventilate the control room. Upon receipt of the actuating signal(s), normal outside air intake supply to the ~~control room~~CRE is isolated, both CRPAR fans are started, the flow path through the Emergency Filtration System is opened, and a portion of the return air volume is filtered to remove airborne contaminants and airborne radioactivity, then mixed with the recirculated return air. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The neutral pressure envelope design of the ~~control room~~CRE minimizes infiltration of unfiltered air from the surrounding areas of the building. The CRPAR System fans are started upon receipt of a safety injection signal or manual initiation through switches in the control room ~~high radiation signal as detected by the radiation monitor R-23 mounted in the main control room emergency zone (CREZ) supply duct~~.

The CRPAR System operation in maintaining a habitable environment in the CRE ~~control room habitable~~ is discussed in the USAR, Section 9.6.4 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers of the CRAC Alternate Cooling System provide double/redundant isolation capability so that the failure of one damper to shut will not result in a breach of control room ventilation isolation. The CRPAR System is designed in accordance with Seismic Category I requirements.

The manual actuation of the CRPAR System during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) is the primary means to ensure CRE habitability in the event of a fuel handling accident while handling recently irradiated fuel. Actuation of the CRPAR System and control room isolation are performed by a SI actuation signal, either automatically or manually initiated. Calculated doses to CRE occupants from a volume control tank rupture or waste gas decay tank rupture are sufficiently small that manual actuation of the CRPAR System is not required for these postulated accidents.

BASES

BACKGROUND (continued)

The CRPAR System is designed to maintain a habitable environment in the CRE ~~control room environment~~ for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem total effective dose equivalent (TEDE).

APPLICABLE SAFETY ANALYSES

The CRPAR System components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the ~~control room envelope~~CRE ensures an adequate supply of filtered air to all areas requiring access. The CRPAR System provides airborne radiological protection for the ~~control room operators~~CRE occupants, as demonstrated by the ~~control room accident~~CRE occupant dose analyses for the most limiting design basis ~~loss of coolant~~ accident~~,~~ fission product release presented in the USAR, Chapter 14 (Ref. 2).

The CRPAR System also provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release (Ref. 6). The evaluation of a smoke challenge also demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panel (Ref. 7).~~for the control room operators in the remote possibility of a fire in the control room, as described in Reference 1.~~

The worst case single active failure of a component of the CRPAR System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

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

LCO

Two independent and redundant CRPAR trains are required to be OPERABLE to ensure that at least one is available ~~assuming~~ if a single active failure disables the other train. Total system failure, such as from a loss of both ventilation trains or from an inoperable CRE boundary, could result in exceeding a dose of 5 rem TEDE to the control room operator in the event of a large radioactive release.

~~The~~ Each CRPAR ~~System~~ train is considered OPERABLE when the individual components necessary to limit ~~operator~~ CRE occupant exposure are OPERABLE~~ in both trains~~. A CRPAR train is OPERABLE when the associated:

a. Fan is OPERABLE;

b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

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

In addition, the CRAC fan in the same train must be OPERABLE when the CRPAR train is required. ~~Furthermore, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.~~

In order for the CRPAR trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

BASES

LCO (continued)

The LCO is modified by ~~a~~ two Notes. The first Note allows~~allowing~~ the ~~control room~~CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, dampers, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE~~control room~~. This individual will have a method to rapidly close the opening and restore the CRE boundary to a condition equivalent to the design condition when a need for ~~control room~~CRE isolation is indicated.

The second Note requires that the CRE be isolated during movement of recently irradiated fuel assemblies. The fuel handling accident analysis assumes the control room is isolated at the initiation of the accident. Pre-isolation of the control room minimizes infiltration of radioactive materials into the CRE prior to initiation of the CRPAR in the emergency mode and ensures dose to CRE occupants remains within applicable limits.

APPLICABILITY

In MODES 1, 2, 3, and 4, ~~5, and 6,~~ and during movement of recently irradiated fuel assemblies, the CRPAR System must be OPERABLE to ensure that the CRE will remain habitable ~~control operator exposure~~ during and following a DBA.

~~In MODE 5 or 6, the CRPAR System is required to cope with the release from the rupture of an inside waste gas tank.~~

During movement of recently irradiated fuel assemblies, the CRPAR System must be OPERABLE to cope with the release from a fuel handling accident involving handling of recently irradiated fuel. The CRPAR is only required to be OPERABLE during fuel handling involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours), due to radioactive decay.

ACTIONS

A.1

When one CRPAR train is inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this condition, the remaining OPERABLE CRPAR train is adequate to perform the ~~control room~~CRE occupant protection function. However, the overall reliability is reduced because a ~~single active~~ failure in the OPERABLE CRPAR train could result in loss of CRPAR function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

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

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem TEDE), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

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

~~If the control room boundary is inoperable in MODE 1, 2, 3, or 4, the CRPAR trains cannot perform their intended functions. Action must be taken to restore an OPERABLE control room boundary within 24 hours. During the period that the control room boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for~~

BASES

ACTIONS (continued)

~~intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the control room boundary.~~

C.1 and C.2

In MODE 1, 2, 3, or 4, if the inoperable CRPAR train or ~~control room~~ the CRE boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1 and D.2

~~In MODE 5 or 6, or d~~During movement of recently irradiated fuel assemblies, if the inoperable CRPAR train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CRPAR train in the emergency mode. This action ensures that the remaining train is OPERABLE and that any active failure would be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the ~~control room~~CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

E.1

~~In MODE 5 or 6, or d~~During movement of recently irradiated fuel assemblies, with two CRPAR trains inoperable, or with one or more CRPAR trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might ~~enter~~ require isolation of the CRE~~control room~~. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

BASES

ACTIONS (continued)

F.1

If both CRPAR trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable ~~control room~~CRE boundary (i.e., Condition B), the CRPAR System may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Operating each CRPAR train for ≥ 15 minutes demonstrates the function of the system. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy~~ availability~~.

SR 3.7.10.2

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

SR 3.7.10.3

This SR verifies that each CRPAR train starts and operates on an actual or simulated actuation (~~high radiation and~~ safety injection) signal. The frequency of 18 months is based on industry operating experience and is consistent with the typical refueling cycle. ~~Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

SR 3.7.10.4

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

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

REFERENCES

1. USAR, Section 9.6.4.

2. USAR, Chapter 14.

3. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI), "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability," dated January 30, 2004. [ADAMS Accession No. ML040300694].

4. Regulatory Guide 1.196, Rev. 2.

5. NEI 99-03, “Control Room Habitability Assessment,” March 2003.

6. Letter from C. R. Steinhardt to NRC, “Submittal of Kewaunee’s Updated Control Room Habitability Evaluation Report to Address Concerns Over Control Room Ventilation,” dated February 28, 1989.

7. USAR Section 9.6.4.

B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Air Conditioning (CRAC) Alternate Cooling System

BASES

BACKGROUND

The CRAC Alternate Cooling System provides temperature control for the control room following isolation of the control room during a design basis accident.

The CRAC Alternate Cooling System consists of two independent and redundant trains that provide cooling of recirculated and fresh air. Each train consists of an air handling unit (AHU) (containing filters, a cooling coil, and a fan), instrumentation, and controls to provide for control room temperature control. The CRAC Alternate Cooling System provides air temperature control for the control room.

The CRAC Alternate Cooling System is an emergency system, parts of which also operate during normal unit operations. A single train will provide the required temperature control to maintain the control room between 60° and 85°F during normal operation using the non-safety related chiller. Under accident conditions (i.e., the non-safety related chillers not in service), cooling from the service water aligned directly to the AHU cooling coils will maintain temperature habitability of the control room environment and will maintain environment temperature for equipment operation. With a service water temperature of 80°F and a 95°F air ambient temperature, each CRAC Alternate Cooling train can maintain control room air temperature within the 110°F design temperature limit. The CRAC Alternate Cooling System operation in maintaining the control room temperature is discussed in the USAR, Section 9.6.4 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the CRAC Alternate Cooling System is to maintain the control room temperature for 30 days of continuous operation.

The CRAC Alternate Cooling System components are arranged in redundant, safety related trains. During emergency operation, the CRAC Alternate Cooling System maintains the temperature < 110°F. A single active failure of a component of the CRAC Alternate Cooling System, with 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. The CRAC Alternate Cooling System is designed in accordance with Nuclear Safety Design Class I requirements. The CRAC Alternate Cooling System 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The CRAC Alternate Cooling System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant trains of the CRAC Alternate Cooling System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disabling the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

The CRAC Alternate Cooling System is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both trains. These components include the cooling coils (with cooling water from the Service Water System) and associated temperature control instrumentation. In addition, the CRAC Alternate Cooling System must be OPERABLE to the extent that air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours), the CRAC Alternate Cooling System must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements following isolation of the control room.

In MODE 5 or 6, CRAC Alternate Cooling System is not required for the mitigation of a postulated event.

ACTIONS

A.1

With one CRAC Alternate Cooling train inoperable, action must be taken to restore OPERABLE status within 30 days. In this condition, the remaining OPERABLE CRAC Alternate Cooling train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE CRAC Alternate Cooling train could result in loss of CRAC Alternate Cooling System function. The 30 day Completion Time is based on the low probability of an event requiring control room isolation, the consideration that the remaining train can provide the required protection, and that alternate safety or nonsafety related cooling means are available.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CRAC Alternate Cooling train cannot be restored to OPERABLE status within the required Completion Time of Condition A, the unit must be placed in a MODE that minimizes the risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed

BASES

ACTIONS

B.1 and B.2 (continued)

Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

During movement of recently irradiated fuel, if the inoperable CRAC Alternate Cooling train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CRAC Alternate Cooling train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE and that active failures will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room (Required Action C.2). This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

During movement of recently irradiated fuel assemblies, with two CRAC Alternate Cooling trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CRAC Alternate Cooling trains are inoperable in MODE 1, 2, 3, or 4, the CRAC Alternate Cooling System may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

BASES ~~(continued)~~

SURVEILLANCE REQUIREMENTS	SR 3.7.11.1 This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the safety analyses in the control room. This SR consists of a combination of testing both redundant cooling units, verifying the availability of cooling water, and calculations. The 18 month Frequency is appropriate since significant degradation of the CRAC Alternate Cooling System is slow and is not expected over this time period.
REFERENCES	1. USAR, Section 9.6.4.

B 3.7 PLANT SYSTEMS

B 3.7.16 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

~~With the specified activity limit, the resultant 2 hour total effective dose equivalent (TEDE) dose to a person at the exclusion area boundary (EAB) would be about 0.03 rem if the steam generator power-operated relief valves (PORVs) open for 2 hours following a trip from full power.~~

Operating a unit at the allowable limits could result in a ~~2 hour~~ EAB exposure of a small fraction of the 10 CFR 50.67 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the USAR, Chapter 14 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of ~~0.10~~ 0.05 μCi/gm 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 an MSLB do not exceed a small fraction of the unit EAB limits (Ref. 1) for TEDE dose ~~rates~~.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 PORVs. The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal (RHR) System to be placed in service. The RHR System then continues to cooldown to 212ºF, at which point the release is terminated.

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 PORVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ ~~0.10~~ 0.05 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

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.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not normally being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

BASES

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies that the secondary specific 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 safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 50.67.

2. USAR, Chapter 14.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

BACKGROUND	A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources - Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of recently irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident involving recently irradiated fuel. Due to radioactive decay, AC electrical power is only required to mitigate fuel handling accidents involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours).

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

This LCO ensures the capability to support systems necessary to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

AC Sources - Shutdown satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems - Shutdown," ensures that all required loads are powered from offsite power. An OPERABLE DG, associated with a distribution system train required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and DG ensures the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling of recently irradiated fuel).

The qualified offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the Engineered Safety Feature (ESF) bus(es).

One qualified offsite circuit consists of the 138/4.16 kV Reserve Auxiliary Transformer, powered by the 138 kV portion of the Kewaunee Substation and normally supplying power to Bus 1-6. The other qualified offsite circuit consists of the 13.8 kV tertiary winding of the 345/138 kV Auto Transformer, powered by either the 345 kV or 138 kV portion of the Kewaunee Substation, to the 13.8/4.16 kV Tertiary Auxiliary Transformer normally supplying power to Bus 1-5. The offsite circuits also include the supply breakers to buses 1-5 and 1-6. While each circuit has connections to each 4.16 kV bus, each circuit is only required to be capable of

BASES

LCO (continued)

supplying one of the 4.16 kV buses at a time. However, if only one offsite circuit is used to meet the LCO requirement, then it must be supplying both buses 1-5 and 1-6.

The DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This sequence must be accomplished within 10 seconds. The DG must be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby at ambient conditions.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY.

It is acceptable for trains to be cross tied during shutdown conditions, allowing a single offsite power circuit to supply all required trains.

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6 and during movement of recently irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;

b. Systems needed to mitigate a fuel handling accident involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. 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 operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

BASES

ACTIONS (continued)

A.1

An offsite circuit would be considered inoperable if it were not available to one required ESF train. Although two trains are required by LCO 3.8.10, the one train with offsite power available may be capable of supporting sufficient required features to allow continuation of recently irradiated fuel movement. By the allowance of the option to declare required features inoperable, with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

A.2.1, A.2.2, A.2.3, B.1, B.2, and B.3

With the offsite circuit not available to all required trains, the option would still exist to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend movement of recently irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC source and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, B.1, B.2, and B.3 (continued)

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power source should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System's ACTIONS would not be entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no AC power to any required ESF bus, the ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit, whether or not a train is de-energized. LCO 3.8.10 would provide the appropriate restrictions for the situation involving a de-energized train.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.9 is not required to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.16 is not required to be met because the ESF actuation signal is not required to be OPERABLE.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude deenergizing a required 4160 V ESF bus or disconnecting a required offsite circuit during performance of SRs. With limited AC sources available, a single event could compromise both the required circuit and the DG. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

REFERENCES

None.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources - Shutdown

BASES

BACKGROUND	A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the USAR, Chapter 14 (Ref. 1), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of recently irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident involving handling of recently irradiated fuel. Due to radioactive decay, DC electrical power is only required to mitigate fuel handling accidents involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours).

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many DBAs have no specific analyses in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The DC Sources - Shutdown satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

One DC electrical power subsystem, consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling within the subsystem, is required to be OPERABLE to support one subsystem of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems - Shutdown." This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (i.e., fuel handling accidents involving handling of recently irradiated fuel and inadvertent dilution events).

APPLICABILITY

The DC electrical power source required to be OPERABLE in MODES 5 and 6, and during movement of recently irradiated fuel assemblies, provides assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;

b. Required features needed to mitigate a fuel handling accident involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) are available;

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

BASES

APPLICABILITY (continued)

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. 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 operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

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

With the required DC electrical power subsystem inoperable, the minimum required DC electrical power subsystem is not available. Therefore, suspension of the movement of recently irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6) is required. Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the Reactor Coolant System (RCS) for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive Moderator Temperature Coefficient (MTC) must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystem and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems.

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystem should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.3. Therefore, see the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

REFERENCES

1. USAR, Chapter 14.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters - Shutdown

BASES

BACKGROUND	A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the USAR, Chapter 14 (Ref. 1), assume Engineered Safety Feature systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protective System and Engineered Safety Features Actuation System instrumentation and controls so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of one inverter to a required 120 VAC instrument bus during MODES 5 and 6 and during movement of recently irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident involving handling of recently irradiated fuel. Due to radioactive decay, DC electrical power is only required to mitigate fuel handling accidents involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours).

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many DBAs have no specific analyses in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The inverters were previously identified as part of the Electrical Power Distribution System and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The required inverter provides uninterruptible supply of AC electrical power to the AC instrument bus even if the 4.16 kV safety buses are de-energized. OPERABILITY of the inverter requires the associated 120 VAC instrument bus to be powered by the inverter with output voltage and frequency within tolerances, and power input to the inverter from a 125 VDC station battery. Power to an instrument bus is provided in the following order: 1) filtered AC through the inverter (referred to as "normal"); 2) DC changed to AC via the inverter (referred to as "standby"); and 3) non-filtered AC through the inverter via a static switch (referred to as "alternate"). Alternatively, power supply may be from an internal AC source via rectifier as long as the station battery is available as the uninterruptible power supply. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (i.e., fuel handling accidents involving handling of recently irradiated fuel and inadvertent dilution events).

BASES

APPLICABILITY

The inverter required to be OPERABLE in MODES 5 and 6 and during movement of recently irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. 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 operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

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

With the required inverter inoperable, suspension of movement of recently irradiated fuel assemblies and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) specified in LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," or boron concentration (MODE 6) specified in LCO 3.9.1, "Boron Concentration," is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive Moderator Temperature Coefficient (MTC) must also be evaluated to ensure they do not result in a loss of required SDM.

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverter and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a constant voltage source transformer.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverter is functioning properly with all required circuit breakers closed and AC instrument bus energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the AC instrument bus. The 7 day Frequency takes into account the other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. USAR, Chapter 14.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems - Shutdown

BASES

BACKGROUND

A description of the AC, DC, and AC instrument bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the USAR, Chapter 14 (Ref. 1), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC, DC, and AC instrument bus 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.

The OPERABILITY of the AC, DC, and AC instrument bus electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC, DC, and AC instrument bus electrical power distribution subsystems during MODES 5 and 6, and during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident involving handling of recently irradiated fuel.

The distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical power distribution system necessary to support OPERABILITY of required systems, equipment, and components - all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

BASES

LCO (continued)

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling of recently irradiated fuel).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of recently irradiated fuel assemblies, provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident involving handling of recently irradiated fuel are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

The AC, DC, and AC instrument bus electrical power distribution subsystems requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. 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 operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of recently irradiated fuel movement. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

distribution subsystem LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend movement of recently irradiated fuel assemblies and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal (RHR) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.3 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR ACTIONS would not be entered. Therefore, Required Action A.2.4 is provided to direct declaring RHR inoperable, which results in taking the appropriate RHR actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.8.10.1 This Surveillance verifies that the AC, DC, and AC instrument bus electrical power distribution subsystems are functioning properly, with all the required buses energized. The verification of proper voltage availability on the required buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these required buses. The 7 day Frequency takes into account the capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.
REFERENCES	1. USAR, Chapter 14.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Containment Penetrations

BASES

BACKGROUND

During movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) within containment, a release of fission product radioactivity within containment will be restricted from escaping 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, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the guidance of Regulatory Guide 1.183 (Ref. 1). Additionally, the containment provides radiation shielding from the fission products that 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 movement of recently irradiated fuel assemblies within containment, the equipment hatch may remain open, but must be capable of being closed ~~must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced~~.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY 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 air lock to remain open for extended periods when frequent containment entry is necessary. During movement of recently irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain capable of being closed.

BASES

BACKGROUND (continued)

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits.

Two systems can be used to purge or ventilate the containment; the Containment Purge and Vent System and the Post LOCA Hydrogen Control System. The Containment Purge and Vent System includes a 36 inch purge penetration and a 36 inch vent penetration. The Post LOCA Hydrogen Control System includes a 2 inch purge penetration and a 2 inch vent penetration. During MODES 1, 2, 3, and 4, the two valves in each of the normal purge and vent penetrations are secured in the closed position. The post LOCA hydrogen control subsystem contains two trains. The valves in Train A are normally closed. The valves in Train B are also normally closed but are periodically opened to control containment pressure within the required limits. The Train B valves receive a signal to close via the Engineered Safety Features Actuation System and the Containment Purge and Vent Isolation System. Neither of the systems are subject to a Specification in MODE 5.

In MODE 6, fresh, tempered air is provided to conduct refueling operations. The normal 36 inch purge system is used for this purpose, and all four valves are closed by the ESFAS in accordance with LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated or capable of being isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during recently irradiated fuel movements.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident involving handling of recently irradiated fuel. The fuel handling accident is a postulated event that involves damage to irradiated fuel. Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly vertically onto a rigid surface or onto other irradiated fuel assemblies. The requirements of LCO 3.9.5, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 100 hours prior to irradiated fuel movement, ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in Regulatory Guide 1.183 (Ref. 1).

BASES

LCO

This LCO limits the consequences of a fuel handling accident involving handling recently irradiated fuel in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except when appropriate administrative controls are in place which ensure the capability to close the penetration ~~for the OPERABLE containment purge and vent penetrations and the containment personnel air locks~~. For the OPERABLE containment purge and vent penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Vent Isolation System.

The LCO is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

The containment personnel air lock doors may be open during movement of recently irradiated fuel in the containment provided that one door is capable of being closed within 30 minutes in the event of a fuel handling accident within containment. When both personnel airlock doors are open during the movement of irradiated fuel in the containment, appropriate plant personnel shall be notified of this condition. A specified individual(s) is designated and available to close the airlock following a required evacuation of containment. Any obstruction(s) (e.g., cables and hoses) that can prevent closure of an open airlock shall be able to be removed in a timely manner (i.e., within the 30 minutes specified above). Should a fuel handling accident occur inside containment, one personnel air lock door will be closed following an evacuation of containment.

The containment equipment hatch may be open during movement of recently irradiated fuel in the containment provided that it is capable of being closed within 45 minutes in the event of a fuel handling accident within containment. When the equipment hatch is open during the movement of irradiated fuel in the containment, appropriate plant personnel shall be notified of this condition. A specified individual(s) is designated and available to close the equipment hatch following a required evacuation of containment. Any obstruction(s) (e.g., cables and hoses) that can prevent closure of the equipment hatch within 45 minutes shall be able to be removed in a timely manner. Should a fuel handling accident occur inside containment, the equipment hatch will be closed following an evacuation of containment.

If it is determined that closure of the equipment hatch and/or containment penetrations would represent a significant radiological hazard to the personnel involved, the decision may be made to forgo closure of the hatch and/or penetrations.

APPLICABILITY

The containment penetration requirements are applicable during movement of recently irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident within containment. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is not being conducted, the potential for a fuel handling accident does not exist. Additionally, due to radioactive decay, a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) will result in doses that are within the guideline values specified in 10 CFR 50.67, even without containment closure capability. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Purge and Vent Isolation System not capable of automatic actuation when the purge and vent valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of recently irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

This Surveillance demonstrates that each required containment penetration is in the required status. The Surveillance on the open purge and vent valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and vent isolation signal.

The Surveillance is performed every 7 days during movement of recently irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident involving handling recently irradiated fuel that releases fission product radioactivity within the containment will not result in a release of significant fission product radioactivity to the environment in excess of those recommended by Regulatory Guide 1.183 (Reference 1).

SR 3.9.6.2

This Surveillance demonstrates that each required containment purge and vent valve actuates to its isolation position on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. LCO 3.3.6, "Containment Purge and Vent Isolation Instrumentation," provides additional Surveillance Requirements for the containment purge and vent valve actuation circuitry. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident involving handling of recently irradiated fuel to limit a release of fission product radioactivity from the containment.

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

REFERENCES

1. Regulatory Guide 1.183, July 2000.
2. USAR, Section 14.2.1.