Progress Energy

TS 5.5.14

Serial: RNP-RA/03-0061

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United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

TRANSMITTAL OF TECHNICAL SPECIFICATIONS BASES REVISIONS

Ladies and Gentlemen:

In accordance with Technical Specifications 5.5.14, Progress Energy Carolinas, Inc., also known as Carolina Power and Light Company, is transmitting revisions to H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, Technical Specifications Bases. The attachment to this letter provides Technical Specifications Bases pages for Revisions 19 through 24.

If you have any questions concerning this matter, please contact me.

Sincerely,

CA Castle
for

C. T. Baucom
Supervisor – Licensing/Regulatory Programs

CAC/cac

Attachment

c: L. A. Reyes, NRC, Region II
NRC Resident Inspector, HBRSEP
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A001

United States Nuclear Regulatory Commission
Attachment to Serial: RNP-RA/03-0061
58 Pages

PROGRESS ENERGY CAROLINAS

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

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BASES

ACTIONS

K.1 and K.2 (continued)

Condition L. The allowance of 48 hours to restore the channel to OPERABLE status, and the additional hour to open the RTBs, are justified in Reference 8.

L.1, L.2, and L.3

Condition L applies when the required number of OPERABLE Source Range Neutron Flux channels is not met in MODE 3, 4, or 5 with the RTBs open. With the unit in this Condition, the NIS source range performs the monitoring and protection functions. With less than the required number of source range channels OPERABLE, operations involving positive reactivity additions shall be suspended immediately. In addition to suspension of positive reactivity additions, all valves that could add unborated water to the RCS must be closed within 1 hour. The isolation of unborated water sources will preclude a boron dilution accident.

Also, the SDM must be verified within 1 hour and once every 12 hours thereafter as per SR 3.1.1.1, SDM verification. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SDM within 1 hour allows sufficient time to perform the calculations and determine that the SDM requirements are met. The SDM must also be verified once per 12 hours thereafter to ensure that the core reactivity has not changed. Required Action L.1 precludes any positive reactivity additions; therefore, core reactivity should not be increasing, and a 12 hour Frequency is adequate. The Completion Times of within 1 hour and once per 12 hours are based on operating experience in performing the Required Actions and the knowledge that unit conditions will change slowly.

Required Action L.1 is modified by a note that permits plant temperature changes provided the temperature change is accounted for in the calculated SDM. Introduction of temperature changes, including temperature increases when a positive MTC exists, must be evaluated to ensure they do not result in a loss of required SDM.

(continued)

BASES

ACTIONS (continued)

E.1

Condition E applies when two hydrogen monitor channels are inoperable. Required Action E.1 requires restoring one hydrogen monitor channel to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable based on the backup capability to sample the containment atmosphere to monitor the hydrogen concentration for evaluation of core damage and to provide information for operator decisions. Also, it is unlikely that a LOCA (which would cause core damage) would occur during this time.

F.1

Condition F applies when the Required Action and associated Completion Time of Condition C, D or E are not met. Required Action F.1 requires entering the appropriate Condition referenced in Table 3.3.3-1 for the channel immediately. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met any Required Action of Condition C, D or E, and the associated Completion Time has expired, Condition F is entered for that channel and provides for transfer to the appropriate subsequent Condition.

G.1 and G.2

If the Required Action and associated Completion Time of Conditions C, D, or E are not met and Table 3.3.3-1 directs entry into Condition G, the unit must be brought to a MODE where the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and MODE 4 within 12 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The containment ventilation isolation radiation monitors ensure closing of the ventilation isolation valves. They are the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits. Due to radioactive decay, containment is only required to isolate during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 56 hours).

The containment ventilation isolation instrumentation satisfies Criterion 3 of the NRC Policy Statement.

LCO

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

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate containment ventilation isolation at any time by using either of two pushbuttons in the control room. Either pushbutton actuates both trains. This action will cause actuation of Phase A and Containment Ventilation Isolation automatic containment isolation valves. Containment Ventilation Isolation can also be initiated by the manual Containment Spray buttons.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one push button and the interconnecting wiring to the actuation logic cabinet.

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays to be OPERABLE. The

(continued)

BASES

LCO

2. Automatic Actuation Logic and Actuation Relays (continued)

Automatic Actuation Logic and Actuation Relays actuate containment ventilation isolation upon receipt of an actuation signal from the Containment Radiation or Manual Initiation Functions. Containment ventilation isolation also initiates on an automatic safety injection (SI) signal when operating in MODES 1, 2, 3, and 4. The Bases for LCO 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," discusses this mode of initiation.

3. Containment Radiation

The LCO specifies two required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment Ventilation Isolation remains OPERABLE.

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

4. Safety Injection

Refer to LCO 3.3.2, Functions 1.a-f, for all initiating Functions and requirements.

APPLICABILITY

The Manual Initiation, Automatic Actuation Logic and Actuation Relays, and Containment Radiation Functions are required to be OPERABLE in MODES 1, 2, 3, and 4, or movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 56 hours) within containment. The Safety Injection Functions are required to be during MODES 1, 2, 3, and 4. Under these conditions, the potential exists for an accident that could release significant fission product radioactivity

(continued)

BASES

APPLICABILITY (continued)

into containment. Therefore, the containment ventilation isolation instrumentation must be OPERABLE in these MODES.

While in MODES 5 and 6 without movement of recently irradiated fuel in progress, the containment ventilation isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

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 are tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1 and A.2

Condition A applies to all Containment Ventilation Isolation Functions and addresses the train orientation of the relay logic and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels. If a train is inoperable or one or more channels are inoperable, operation may continue as long as the Required Action to place and maintain containment purge supply and exhaust isolation valves in their closed

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The radiation monitor actuation of the CREFS during movement of irradiated fuel assemblies is the primary means to ensure control room habitability in the event of a fuel handling or waste gas decay tank rupture accident.

The CREFS actuation instrumentation satisfies Criterion 3 of the NRC Policy Statement.

LCO

The LCO requirements ensure that instrumentation necessary to initiate the CREFS 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. The applicable MODES and specified conditions for the CREFS portion of these functions are different and less restrictive than those specified for their SI roles. If one or more of the SI functions becomes inoperable in such a manner that only the CREFS function is affected, the Conditions applicable to their SI function need not be entered. The less restrictive Actions specified for inoperability of the CREFS Functions specify sufficient compensatory measures for this case.

2. Control Room Radiation Monitor

The LCO requires one Control Room Area Radiation Monitor OPERABLE to initiate the CREFS.

3. Safety Injection

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

(continued)

BASES

APPLICABILITY

The CREFS Functions must be OPERABLE in MODES 1, 2, 3, 4, and during movement of irradiated fuel assemblies. Applicability to movement of irradiated fuel excludes movement of irradiated fuel within a properly sealed spent fuel shipping cask.

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 Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.7-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function are tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the automatic actuation Function of the CREFS.

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 3.7.9. If the channel/train cannot be restored to OPERABLE status, one CREFS train must be placed in the emergency pressurization mode of operation. This accomplishes the actuation instrumentation Function and places the unit in a conservative mode of operation.

(continued)

BASES

ACTIONS (continued)

B.1

Condition B applies to the failure of two CREFS actuation trains, or the radiation monitor channel. The Required Action is to place one CREFS train in the emergency pressurization 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.

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 irradiated fuel assemblies are being moved. Movement of irradiated fuel assemblies must be suspended immediately to reduce the risk of accidents that would require CREFS actuation.

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 CREFS Actuation Functions.

SR 3.3.7.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of radiation monitor instrumentation has not occurred.

(continued)

BASES (continued)

LCO (continued)

tests that are designed to validate various accident analyses values. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. Another test performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow.

The no flow test may be performed in MODE 3 or 5 and requires that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should be performed only once unless the flow characteristics of the RCS are changed. The 1 hour time period specified is adequate to perform the desired tests, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure the SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

c. The Rod Control System is not capable of rod withdrawal, the reactor trip breakers are open, or the lift disconnect switches for all control rods not fully withdrawn are open. Any of these conditions

(continued)

BASES

LCO (continued)

loops or trains that are required to be OPERABLE to consist of any combination of RCS loops and RHR trains. Any one loop or train in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop or train is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs or RHR pumps to be de-energized for ≤ 1 hour in any 8 hour period. The purpose of the Note is to permit tests that are designed to validate various accident analyses values.

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, therefore maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure the SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

c. The Rod Control System is not capable of rod withdrawal, due to the postulation of a power excursion because of an inadvertent control rod withdrawal.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.16.3 (continued)

15 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates the $\bar{E}$ determination is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10 CFR 100.11, 1973.

2. UFSAR, Section 15.6.3.

BASES (continued)

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The inboard 42 inch purge valves must have blocks installed to prevent full opening and actuate closed on an automatic signal. The valves covered by this LCO are listed along with their associated stroke times in the Inservice Testing Program.

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact.

This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

level of 0% with the single failure of a steam line check valve for the most limiting pressure response and 102% of the pre-Appendix K power uprate licensed power level of 2300 MWt (i.e., 2346 MWt) with the single failure of an emergency bus for the most limiting temperature response. Both analyses assume initial (pre-accident) containment conditions of 130°F and 1.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 2).

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a -3.0 psig containment pressure and is associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-High pressure setpoint to achieving full flow through the containment spray nozzles.

Containment cooling train performance for post accident conditions is given in Reference 3. The result of the analysis is that each train can provide 100% of the required peak cooling capacity during the post accident condition. The train post accident cooling capacity under varying containment ambient conditions, is also shown in Reference 4. The modeled Containment Cooling System actuation from the containment analysis is based on a response time associated with exceeding the containment high pressure setpoint to achieving full Containment Cooling System air and cooling water flow.

The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.7

This SR requires verification that each containment cooling train actuates upon receipt of an actual or simulated safety injection signal. The 18 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the 18 month Frequency.

SR 3.6.6.8

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Performance is required following activities which could result in nozzle blockage. Such activities may include: (1) a major configuration change; or (2) a loss of foreign material control such that the final condition of the system cannot be assured. The frequency is considered adequate due to the passive design of the nozzles, the stainless steel construction of the piping and nozzles, and the use of foreign material exclusion controls during system opening.

REFERENCES

1. UFSAR, Section 3.1.
2. 10 CFR 50, Appendix K.
3. UFSAR, Section 6.2.
4. UFSAR, Section 9.4.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

BASES

APPLICABLE SAFETY ANALYSES (continued)

event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value. When the Moderator Temperature Coefficient (MTC) is positive, the reactor power may increase above the initial value during an RCS heatup event (e.g., turbine trip). Thus, for any number of inoperable MSSVs it is necessary to reduce the trip setpoint if a positive MTC may exist at partial power conditions.

The MSSVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The accident analysis assumes four MSSVs per steam generator are OPERABLE to provide overpressure protection for design basis transients occurring at 102% of the pre-Appendix K power uprate licensed power level of 2300 MWt (i.e., 2346 MWt). The LCO, therefore, also requires that four MSSVs per steam generator be OPERABLE.

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB, or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, four MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

reverse flow in the case of a steam line break in that line. Due to the presence of check valves, the loss of one MSIV was not analyzed.

The limiting case for containment pressure response is the SLB inside containment at hot zero power, with single failure of one steam line check valve. This case releases the largest integrated mass into containment. The pressure rise is very steep initially, then moderates as the break flow rate decreases. The limiting case for containment temperature response is the SLB inside containment from 102% of the pre-Appendix K power uprate power level of 2300 MWt (i.e., 2346 MWt), with single failure of an emergency bus, and without entrainment. This case maximizes the integrated energy deposited into the containment during the early portion of the event. Blowdown fluid enthalpies allow the steam entering the containment to remain superheated. When the containment sprays actuate, the superheated steam is rapidly condensed, and the temperature quickly falls to the saturation temperature at the partial pressure of the steam.

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at hot zero power with offsite power available is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available, and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System cooldown. With a loss of offsite power, the response of mitigating systems is delayed. Significant single failures considered include loss of one safety injection pump.

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. A steam line break causes a main steam isolation signal to be generated by either high steam flow coincident with low T_{avg} or with low steam pressure, or high-high containment pressure. This action prevents

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFRV and bypass valve is ≤ 20 seconds on an actual or simulated actuation signal. The MFRV, and bypass valve closure times are assumed in the accident and containment analyses (Ref. 2). This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code, Section XI (Ref. 3).

The Frequency for this SR is in accordance with the Inservice Testing Program. The specified Frequency for valve closure is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the specified Frequency.

SR 3.7.3.2

This SR verifies that the closure time of each MFIV is ≤ 50 seconds on an actual or simulated actuation signal. The MFIV closure times are assumed in the accident and containment analyses (Ref. 2). This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code, Section XI (Ref. 3).

The Frequency for this SR is in accordance with the Inservice Testing Program. The specified Frequency for valve closure is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the specified Frequency.

(continued)

BASES

ACTIONS (continued)

E.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With one required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

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

SR 3.7.4.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 4) to monitor centrifugal pump performance. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. This ensures that pump performance is consistent with the pump curve. Performance of inservice testing discussed in the ASME Code, Section XI (Ref. 4)

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.2 (continued)

(only required at 3 month intervals) satisfies this requirement. The 31 day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 4.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.4.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an AFW actuation signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

This SR is modified by a Note that states the SR is not required in MODE 4 when AFW is being used for heat removal. In MODE 4, the required AFW train is already aligned and operating.

SR 3.7.4.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an AFW actuation

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.4.6

This SR verifies that the automatic bus transfer switch associated with the "swing" motor driven AFW flow path discharge valve V2-16A will function properly to automatically transfer the power source from the aligned emergency power source to the other emergency power source upon loss of power to the aligned emergency power source. The Surveillance consists of two tests to assure that the switch will perform in either direction. One test is performed with the automatic bus transfer switch aligned to one emergency power source initially, and the test is repeated with the switch initially aligned to the other emergency power source. Periodic testing of the switch is necessary to demonstrate OPERABILITY. Operating experience has shown that this component usually passes the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.4.8.
2. UFSAR, Section 15.2.8.
3. UFSAR, Section 15.2.7.
4. ASME, Boiler and Pressure Vessel Code, Section XI.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting event for the condensate volume is the loss of offsite power because of the loss of makeup capability to the CST. A backup water supply to feed the steam generators is provided through a direct connection between the Service Water System (SWS) and the AFW System.

The CST satisfies Criterion 3 of the NRC Policy Statement.

LCO

To satisfy operational requirements, the CST must contain sufficient cooling water to remove decay heat for 2 hours following a reactor trip from 102% of the pre-Appendix K power uprate licensed power level of 2300 MWt (i.e., 2346 MWt), assuming a coincident loss of offsite power and the most adverse single active failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps.

The CST level required is equivalent to a usable volume of ≥ 35,000 gallons, which is based on holding the unit in MODE 3 for 2 hours.

The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level.

The backup SWS supply to the AFW System must also be OPERABLE to satisfy decay heat removal requirements in the event of a loss of normal make-up capability to the CST resulting from a loss of offsite power.

APPLICABILITY

In MODES 1, 2, 3, and in MODE 4, when a steam generator is being used for heat removal, the CST is required to be OPERABLE.

In MODE 5 or 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the CST level is not within limits, the OPERABILITY of the backup supply should be verified by administrative means within 4 hours and once every 12 hours thereafter.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The UHS satisfies Criterion 3 of the NRC Policy Statement.

LCO

The UHS is required to be OPERABLE and is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate for at least 22 days following the design basis LOCA without the loss of NPSH, and without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the UHS temperature should not exceed 97°F and the level should not fall below 218 ft MSL during normal unit operation.

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

A.1 and A.2

With the SW temperature > 97°F but ≤ 99°F, the required cooling capacity of the SW System must be verified by evaluating the existing operational condition of the systems and components served by the SW System and verifying that each is capable of performing its safety related function. The required cooling capacity must also be re-verified once per 12 hours. In addition, the SW temperature must be verified ≤ 99°F once per hour. The temperature verification ensures the SW temperature remains below the maximum water temperature allowed for the safety related components to perform their safety function.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

The Completion Time of Required Action A.1 was developed considering that some activities required to complete the evaluation of required cooling capacity could be completed prior to the Condition being entered.

The Completion Time of Required Action A.2 is based on shift schedules for convenience and is considered acceptable since temperature monitoring capability is available to detect an increase in SW temperature throughout the period of Condition A.

B.1 and B.2

If the Required Actions and associated Completion Times are not met or the UHS is inoperable for reasons other than Condition A, 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 3.7.8.1

This SR verifies that adequate long term (22 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the SWS pumps. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is $\geq$ 218 ft MSL.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.2

This SR verifies that the SWS is available to cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the service water temperature is $\leq$ 97°F.

REFERENCES

1. UFSAR, Section 9.2.4.

2. UFSAR Section 2.4.6.1.

3. UFSAR Section 2.1.1.2.

4. NUREG-75/024, "Final Environmental Statement Related to the Operation of H. B. Robinson Nuclear Steam-Electric Plant Unit 2," U. S. Nuclear Regulatory Commission, Washington DC 20555, April 1975, page 3-7.

5. USGS Historical Daily Values for Station Number 02130900, Black Creek Near McBee, South Carolina, Years 1960-1993.

BASES

BACKGROUND (continued)

the control room envelope. Periodic testing is required to demonstrate that the control room is pressurized to a minimum of 0.125 inches water gage with respect to the outdoors, and to a positive pressure with respect to adjacent areas, with an outside air makeup rate of ≤ 400 CFM, while in the emergency pressurization mode of operation. Periodic testing also demonstrates that a positive pressure can be maintained in the control room with respect to the outdoors. The CREFS operation in maintaining the control room habitable is discussed in the Updated Final Safety Analysis Report (UFSAR), Section 6.4 (Ref. 1).

Pressurization of the control room habitability envelope by the CREFS assumes that non-safety related ventilation fans in the Auxiliary Building adjacent to the control room either remain in operation or cease operation. In the event that the air supply fan to the Auxiliary Building remains in operation simultaneously with the Auxiliary Building air exhaust fan not in operation, one room of the Auxiliary Building (i.e., Hagan Room) could be slightly positive with respect to the control room. Procedures require that the air supply fan to the Auxiliary Building be shut down within one hour of actuation of the CREFS to assure that the air pressure in the Auxiliary Building is reduced. Analyses show that the dose to the control room operator is satisfactory under this condition (Ref. 2).

The air entering the control room through the outside air intake is continuously monitored for radiation in the control room and smoke in the ventilation air duct.

The CREFS is designed to maintain the control room environment for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem whole body dose or its equivalent to any part of the body, or 5 rem total effective dose equivalent (TEDE) for a fuel handling accident.

APPLICABLE SAFETY ANALYSES

The active CREFS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The CREFS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident, fission

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

product release presented in the UFSAR, Chapter 15 (Ref. 3).

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

The CREFS satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two redundant CREFS trains are required to be OPERABLE to ensure that at least one is available assuming a single active failure disables the other train. Total system failure could result in exceeding a dose of 5 rem whole body or its equivalent to any part of the body to the control room operator in the event of a large radioactive release, or 5 rem TEDE for a fuel handling accident.

The CREFS is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in both trains. A CREFS train is OPERABLE when the air cleaning unit fan, air recirculation fan, air intake damper and associated ductwork, and air exhaust damper and associated ductwork, are operable for the given train. The common air filtration unit is OPERABLE to support either train in accordance with the Ventilation Filter Testing Program. In addition, non-redundant ductwork and gravity dampers are OPERABLE to support either train. Implicit in the OPERABILITY of either train is that the integrity of the control room envelope is such that it can be pressurized to ≥0.125" water gauge relative to the outside atmosphere and to a positive pressure relative to adjacent areas at a make-up rate of ≤400 cfm in the emergency pressurization mode.

APPLICABILITY

In MODES 1, 2, 3, 4, and during movement of irradiated fuel assemblies, CREFS must be OPERABLE to control operator exposure during and following a DBA. During movement of irradiated fuel assemblies, the CREFS must be OPERABLE to cope with the release from a fuel handling accident. Applicability to movement of irradiated fuel excludes movement of irradiated fuel within a properly sealed spent fuel shipping cask.

(continued)

BASES

ACTIONS

A.1

When one CREFS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREFS train is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CREFS train could result in loss of CREFS 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 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CREFS train 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.

C.1 and C.2

During movement of irradiated fuel assemblies, if the inoperable CREFS train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREFS train in the emergency pressurization mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action C.1 is to immediately 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.

(continued)

BASES

ACTIONS (continued)

D.1

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

E.1

If both CREFS trains are inoperable in MODE 1, 2, 3, or 4, action must be taken to restore OPERABLE status of at least one CREFS train within 48 hours. The 48 hour completion time is based upon the low probability of a DBA occurring during this time.

F.1 and F.2

In MODE 1, 2, 3, or 4, if both inoperable CREFS trains 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.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

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. Operation for ≥ 15 minutes is adequate to demonstrate the function of the system. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy availability.

(continued)

BASES

LCO

Two independent and redundant trains of the CREAC WCCUs 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.

A WCCU train is OPERABLE when the refrigeration equipment of a particular train is capable of removing the design heat load. Implicit in the operability of the WCCU trains are the instrumentation and controls necessary to support automatic start and temperature control operation. Also implicit in the operability of the WCCU trains is the operability of the SWS supply to the WCCU subsystem.

APPLICABILITY

In MODES 1, 2, 3, 4, and during movement of irradiated fuel assemblies, the WCCUs must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements. Applicability to movement of irradiated fuel excludes movement of irradiated fuel within a properly sealed spent fuel shipping cask.

ACTIONS

A.1

With one WCCU train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE WCCU train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE WCCU train could result in loss of cooling function. The 30 day Completion Time is based on the consideration that the remaining train can provide the required cooling.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable WCCU train cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes the risk. To achieve this status, the unit must

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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.

C.1 and C.2

During movement of irradiated fuel, if the inoperable WCCU train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE WCCU train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, 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 emergency pressurization of the control room. 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 irradiated fuel assemblies, with two WCCU 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 WCCU trains are inoperable in MODE 1, 2, 3, or 4, action must be taken to restore at least one WCCU train to OPERABLE status within 48 hours. The 48 hour completion time is based upon the low probability of a Design Basis Accident occurring during this time.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.11 Fuel Building Air Cleanup System (FBACS)

BASES

BACKGROUND

The FBACS filters airborne radioactive particulates from the area of the spent fuel pool following a fuel handling accident in the Fuel Building. The FBACS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the spent fuel pool area.

The FBACS is a single train system which consists of a heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system.

The FBACS is a manually initiated system, which may also be operated during normal plant operations.

The FBACS is discussed in the UFSAR, Sections 6.5.1, 9.4.5, and 15.7.4 (Refs. 1, 2, and 3, respectively) because it may be used for normal, as well as post accident, atmospheric cleanup functions.

APPLICABLE SAFETY ANALYSES

The FBACS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident in the Fuel Building. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged and the fission product inventory in the gap is released. The FBACS is assumed to be operating during the release and a once through filter efficiency of 90% for elemental iodine and 70% for organic iodine is assumed. All of the release passes through the FBACS due to the negative air pressure maintained by the FBACS in the Fuel Building, (i.e., no bypass leakage is assumed). The integrated dose is calculated using assumptions in Reference 3, which are consistent with the methodology utilized

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

in Regulatory Guide 1.183 (Ref. 8).

The FBACS satisfies Criterion 3 of the NRC Policy Statement.

LCO

The FBACS is required to be OPERABLE and operating. Total system failure could result in the atmospheric release from the fuel handling building exceeding the 10 CFR 50.67 (Ref. 4) limits in the event of a fuel handling accident.

The FBACS is considered OPERABLE when the individual components necessary to control exposure in the fuel handling building are OPERABLE. The FBACS is considered OPERABLE when its:

a. Fan is OPERABLE;

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

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

APPLICABILITY

During movement of irradiated fuel in the fuel handling area, the FBACS is required to be OPERABLE and operating to alleviate the consequences of a fuel handling accident.

ACTIONS

A.1

When the FBACS is inoperable during movement of irradiated fuel assemblies in the fuel building, action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.11.3 (continued)

permit cask egress, ISTS SR 3.7.13.4 cannot be met. OPERABILITY of the FBACS is not necessary when irradiated fuel assemblies are in a spent fuel shipping cask because irradiated fuel assemblies are protected from damage and associated release of fission products by the cask and other controls associated with shipments of spent fuel assemblies.

REFERENCES

1. UFSAR, Section 6.5.1.
2. UFSAR, Section 9.4.5.
3. UFSAR, Section 15.7.4.
4. 10 CFR 50.67.
5. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.
6. Licensee Event Report (LER) 50-26/97-05, dated May 22, 1997.
7. Certificate of Compliance No. 9001 for Spent Fuel Shipping Cask Model IF-300, dated March 26, 1996.
8. Regulatory Guide 1.183.

B 3.7 PLANT SYSTEMS

B 3.7.12 Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level of 21 ft above the top of the fuel in the fuel storage pool exceeds the assumptions of iodine decontamination factors following a fuel handling accident and bounds the sensible heat sink assumptions used in "time to boil" calculations. With the fuel storage racks installed in the spent fuel storage pool, a water level 21 ft above the fuel corresponds to approximately 35 ft pool water depth. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the UFSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 15.7.4 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Reference 3. The resultant 2 hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 50.67 (Ref. 4) limits.

According to the fuel storage pool fuel handling accident analysis (Ref. 3), the minimum level of 21 ft over the top of irradiated fuel assemblies seated in the storage racks exceeds the submergence requirements necessary to obtain the assumed decontamination factor (DF) for inorganic iodines released from damaged fuel as a result of the accident.

The fuel storage pool water level satisfies Criterion 2 of the NRC Policy Statement.

LCO

The fuel storage pool water level is required to be ≥ 21 ft over the top of irradiated fuel assemblies seated in the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1 (continued)

the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

During fuel transfer operations, the level in the fuel storage pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.6.1.

REFERENCES

1. UFSAR, Section 9.1.2.
2. UFSAR, Section 9.1.3.
3. UFSAR, Section 15.7.4.
4. 10 CFR 50.67.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During movement of recently irradiated fuel assemblies 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 requirements of 10 CFR 100. 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 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 lock, which is also part of the containment pressure boundary, provides a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Lock." The 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

(continued)

BASES

BACKGROUND (continued)

when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of the 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 closed.

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident involving handling recently irradiated fuel during refueling.

The Containment Ventilation System includes the Containment Purge System and the Containment Pressure and Vacuum Relief System. The Containment Purge System has a 42 inch supply penetration and a 42 inch exhaust penetration. The Containment Pressure and Vacuum Relief System has two separate 6 inch penetrations. The two valves in each of the penetrations can be opened intermittently, but are closed automatically by the Containment Isolation System. Neither of the subsystems is subject to a Specification in MODE 5.

In MODE 6, large air exchanges are necessary to conduct refueling operations. The normal 42 inch purge system is used for this purpose, and all four isolation valves are automatically closed in accordance with LCO 3.3.6, "Containment Ventilation Isolation Instrumentation." The Containment Pressure and Vacuum Relief System remains operational in MODE 6, and all four isolation valves are also automatically closed by the Containment Ventilation Isolation System.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be 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 fuel movements.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident involving handling recently irradiated fuel. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents analyzed include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Cavity Water Level," and irradiated fuel movement with containment closure capability or a minimum decay time of 56 hours without containment closure capability ensure 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 10 CFR 50.67. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

Containment penetrations satisfy Criterion 3 of the NRC Policy Statement.

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 for the OPERABLE containment ventilation penetrations. For the OPERABLE containment ventilation penetrations, this LCO ensures that these penetrations are isolable by the Containment Ventilation Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic containment ventilation valve closure times specified in the UFSAR can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.

APPLICABILITY

The containment penetration requirements are applicable during movement of recently irradiated fuel assemblies within containment because this is when there is a potential

(continued)

BASES (continued)

APPLICABILITY (continued)

for the limitting fuel handling accident. 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 fuel that was not recently irradiated (i.e., fuel that has not occupied part of a critical reactor core within the previous 56 hours) will result in doses that are well 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 lock, 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 Ventilation Isolation System not capable of automatic actuation when the containment ventilation 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.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open ventilation 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 ventilation isolation signal.

The Surveillance is performed every 7 days during movement of recently irradiated fuel assemblies within containment. This Surveillance ensures that a postulated fuel handling

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1 (continued)

Accident involving handling recently irradiated fuel that releases fission product radioactivity within the containment will not result in a significant release of fission product radioactivity to the environment.

SR 3.9.3.2

This Surveillance demonstrates that each containment ventilation valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar instrumentation and valve testing requirements. In LCO 3.3.6, the Containment Ventilation Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident involving handling recently irradiated fuel to limit a release of fission product radioactivity from the containment.

REFERENCES

1. UFSAR, Section 15.7.4.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Ref. 1). Sufficient iodine activity would be retained to limit offsite doses from the accident to within Regulatory Guide 1.183 and 10 CFR 50.67 limits (Refs. 2 and 3).

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 200 to be used in the accident analysis for iodine. Therefore, consistent with Regulatory Guide 1.183, Appendix B.2, the overall effective iodine decontamination factor is 200 for the refueling cavity, with a resulting chemical species released from the water of 57% elemental and 43% organic iodine (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 1. With a minimum water level of 23 ft and a minimum decay time of 56 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Refs. 2 and 3).

Refueling cavity water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits.

APPLICABILITY

LCO 3.9.6 is applicable when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.12, "Fuel Storage Pool Water Level."

ACTIONS

A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 1).

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1 (continued)

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. UFSAR, Section 15.7.4.
2. 10 CFR 50.67.
3. Regulatory Guide 1.183.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Containment Purge Filter System

BASES

BACKGROUND

The Containment Purge Filter System filters airborne radioactivity released to the containment atmosphere following a fuel handling accident involving handling recently irradiated fuel in the containment. During refueling outages, the Containment Purge Filter System, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the containment.

The Containment Purge Filter System is a single train system which consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and two fans (only one of the fans is required, the second fan is a spare). Ductwork, valves or dampers, and instrumentation also form part of the system.

The Containment Purge Filter System is a manually intitiated system, which may also be operated during normal plant operations.

The Containment Purge Filter System is discussed in the UFSAR, Sections 6.5.1, 9.4.3, and 15.7.4 (Refs. 1, 2, and 3, respectively) because it may be used for normal, as well as post accident, atmospheric cleanup functions.

APPLICABLE SAFETY ANALYSES

The containment purge filter system is not used for mitigation of the fuel handling accident as described in UFSAR Section 15.7.4. This system is required to be OPERABLE and in operation during the movement of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 56 hours). In the event of a fuel handling accident involving recently irradiated fuel, the containment purge filter system, in conjunction with the containment ventilation isolation requirements of LCO 3.3.6 and the containment closure requirements of LCO 3.9.3, would significantly impede the radioactive release.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Containment Purge Filter System satisfies Criterion 3 of the NRC Policy Statement.

LCO

The Containment Purge Filter System is required to be OPERABLE and operating. When the Containment Purge Filter System is in operation, the exhaust flow from containment shall discharge through the HEPA and impregnated charcoal filters.

The Containment Purge Filter System is considered OPERABLE when:

a. One fan is OPERABLE;

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

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

APPLICABILITY

During movement of recently irradiated fuel in the containment, the Containment Purge Filter System is required to be OPERABLE and operating to alleviate the consequences of a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 56 hours).

ACTIONS

A.1 and A.2

When the Containment Purge Filter System is inoperable or not in operation during movement of recently irradiated fuel assemblies in containment, Required Action A.1 requires each penetration providing direct access from the containment atmosphere to the outside atmosphere to be immediately

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

closed. Closure may be achieved by a closed manual or automatic valve, blind flange, or equivalent method. Equivalent closure methods must be approved and may include use of a material that can provide a temporary atmospheric pressure, ventilation barrier for the penetration during fuel movements. Alternately, Required Action A.2 may be taken to place the unit in a condition in which the LCO does not apply. Required Action A.2 requires immediate suspension of movement of recently irradiated fuel assemblies in containment. Suspension of this activity does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

This SR verifies that the relative humidity of the containment atmosphere to be processed by the Containment Purge Filter System is ≤ 70%. This ensures that the testing performed to validate the safety analysis assumptions relative to charcoal filter efficiency, bounds actual plant conditions for relative humidity at the inlet of the Containment Purge Filter System charcoal filter. The one hour Frequency is based on engineering judgment considering the likelihood of changes in containment relative humidity during refueling outages.

SR 3.9.7.2

This SR verifies that the Containment Purge Filter System is in operation and maintaining containment pressure negative relative to the adjacent auxiliary building areas once every 12 hours. This verification ensures that containment pressure is being maintained negative with respect to the outside atmosphere since the pressure of the auxiliary building areas is normally maintained negative with respect to the outside atmosphere. The Containment Purge Filter

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.7.2 (continued)

System is assumed to maintain a slight negative pressure in the containment, to prevent unfiltered leakage to the outside atmosphere. The Frequency of 12 hours is sufficient considering other indications available to the operator to monitor Containment Purge Filter System operation.

SR 3.9.7.3

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

REFERENCES

1. UFSAR, Section 6.5.1.
2. UFSAR, Section 9.4.3.
3. UFSAR, Section 15.7.4.
4. 10 CFR 50.67.