April 8, 1999

Mr. W. R. McCollum, Jr.
Vice President, Oconee Site
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P. O. Box 1439
Seneca, SC 29679

SUBJECT: TECHNICAL SPECIFICATION BASES CHANGES - OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

Dear Mr. McCollum:

By letter dated April 1, 1999, you informed the staff of changes to the Oconee Nuclear Station, Units 1, 2, and 3 Technical Specifications (TS) that have been implemented on March 27, 1999, concurrent with implementation of the Improved TS (ITS). The Bases affected are for Sections 2.1.1, 3.1.1, 3.1.3, 3.3.1, 3.3.5, 3.3.7, 3.3.17, 3.3.22, 3.4.12, 3.5.1, 3.5.2, 3.5.3, 3.6.1, 3.6.2, 3.6.3, 3.6.4, 3.7.5, 3.7.8, 3.7.17, 3.8.1, 3.8.2, and 3.10.1.

The purpose of the changes is to either make the Bases more consistent with the discussion of changes in the ITS submittal and supplements or provide corrections and enhancements to the ITS Bases.

The purpose of this letter is to distribute the attached revised TS pages to the appropriate TS manual holders.

Sincerely,
original signed by:
David E. LaBarge, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosure: Bases Change

cc w/encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Enclosure: Bases Change

cc w/encl: See next page

Oconee Nuclear Station

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TABLE OF CONTENTS

(continued)

BASES (continued)

APPLICABILITY

SL 2.1.1.1 and SL 2.1.1.2 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The MSRVs, or automatic protection actions, serve to prevent RCS heatup to reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.

In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the requirement to go to MODE 3 places the unit in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

REFERENCES

1. UFSAR, Section 3.1.
2. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," April 1995.
3. UFSAR, Chapter 15.

BASES

BACKGROUND (continued)

of a main steam line break (MSLB)in MODE 3, 4, or 5 when high steam generator levels exist.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and anticipated transients, with assumption of the highest worth CONTROL ROD stuck out following a reactor trip.

The criteria for SDM requirements are that specified acceptable fuel design limits are maintained. The SDM requirements must ensure that:

a. The reactor can be made subcritical from all operating conditions, transients, and other Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable with acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for anticipated transients, and ≤ 280 cal/gm energy deposition for the rod ejection accident (Ref. 3)); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements is based on an MSLB, as described in the accident analyses (Ref. 2). In addition to the limiting MSLB accident, the SDM requirement must also protect against other accidents described in UFSAR Chapter 15 (Ref. 2).

The basis for the shutdown requirement when high steam generator levels exist is the heat removal potential in the secondary system fluid and the positive reactivity added via MTC. At any given initial primary system temperature and its associated secondary system pressure, the secondary system liquid levels can be equated to a final primary system temperature assuming the entire secondary system mass is boiled away or reaches the thermal equilibrium with the primary system. A SDM at 70°F with the highest worth rod

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

stuck out will bound all resulting SDM's in the event that the entire secondary system is boiled away. However, a 200°F SDM is adequately conservative since the RCS will not cool down below this temperature in the event of a MSLB with raised SG levels.

SDM satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

Shutdown boron concentration requirements assume the highest worth CONTROL ROD is stuck in the fully withdrawn position to account for a postulated inoperable or untrippable rod prior to reactor shutdown.

SDM is a core design condition that can be ensured through CONTROL ROD positioning and through the soluble boron concentration.

The MSLB (Ref. 2) accident is the most limiting analysis that establishes the SDM value of the LCO.

For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100 limits (Ref. 5).

To compensate for the potential heat removal associated with an MSLB accident when high steam generator levels exist, such as during secondary system chemistry control and steam generator cleaning, the initial SDM in the core must be adjusted. The operating procedures provide adjusted SDM limits that ensure the core will remain subcritical following an MSLB accident from those conditions.

APPLICABILITY

In MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analysis discussed above. The operating procedures are used to define the SDM when high steam generator levels exist, such as during secondary system chemistry control and steam generator cleaning in MODES 3, 4, and 5. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5 and LCO 3.2.1. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

MTC values are bounded in reload safety evaluations, assuming steady state conditions at BOC and EOC.

MTC satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

LCO 3.1.3 requires the MTC to be within specified limits in the COLR to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the accident analysis during operation. The LCO establishes a maximum positive value that can not be exceeded. The limit in the COLR on positive MTC ensures that core overheating accidents will not violate the accident analysis assumptions. The limit on MTC when THERMAL POWER is ≥ 95% RTP, ensures that steady state core operation will be stable.

MTC is a core physics parameter determined by the fuel and fuel cycle design and cannot be controlled directly once the core design is fixed during operation; therefore, the LCO can only be ensured through measurement. The surveillance check at BOC on MTC provides confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met.

APPLICABILITY

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure that startup and subcritical accidents, such as the uncontrolled CONTROL ROD group withdrawal, will not violate the assumptions of the accident analysis. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no accidents using the MTC as an analysis assumption are initiated from these MODES. However, the variation of MTC with temperature in MODES 3, 4, and 5 for accidents initiated in MODES 1 and 2 is accounted for in the subject accident analysis. The Surveillance check at BOC on MTC provides confirmation that MTC is behaving as anticipated.

(continued)

BASES

BACKGROUND

Channel Bypass (continued)

alarms/indicator lights. This is administratively controlled by having only one manual bypass key available for each unit. All RPS trips are reduced to a two-out-of-three logic in channel bypass.

Shutdown Bypass

During unit cooldown and heatup, it is desirable to leave the safety rods at least partially withdrawn to provide shutdown capabilities in the event of unusual positive reactivity additions (moderator dilution, etc.).

However, the unit is also depressurized as coolant temperature is decreased. If the safety rods are withdrawn and coolant pressure is decreased, an RCS Low Pressure trip will occur at 1800 psig and the rods will fall into the core. To avoid this, the protective system allows the operator to bypass the low pressure trip and maintain shutdown capabilities. During the cooldown and depressurization, the safety rods are inserted prior to the low pressure trip of 1800 psig. The RCS pressure is decreased to less than 1720 psig, then each RPS channel is placed in shutdown bypass.

In shutdown bypass, a normally closed contact opens when the operator closes the shutdown bypass key switch (status shall be indicated by a light). This action bypasses the RCS Low Pressure trip, Nuclear Overpower Flux/Flow Imbalance trip, Reactor Coolant Pump to Power trip, and the RCS Variable Low Pressure trip, and inserts a new RCS High Pressure, 1720 psig trip. The operator can now withdraw the safety rods for additional rapidly insertable negative reactivity.

The insertion of the new high pressure trip performs two functions. First, with a trip setpoint of 1720 psig, the bistable prevents operation at normal system pressure, 2155 psig, with a portion of the RPS bypassed. The second function is to ensure that the bypass is removed prior to normal operation. When the RCS pressure is increased during a unit heatup, the safety rods are inserted prior to reaching 1720 psig. The shutdown bypass is removed, which returns the RPS to normal, and system pressure is increased

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Keowee Hydro Unit Emergency Start

The ESPS initiated Keowee Hydro Unit Emergency Start has been included in the design to ensure that emergency power is available throughout the limiting LOCA scenarios.

The small break LOCA analyses assume a conservative 48 second delay time for the actuation of HPI and LPI in UFSAR, Chapter 15 (Ref. 4). The large break LOCA analyses assume LPI flow starts in 38 seconds while full LPI flow does not occur until 15 seconds later, or 53 seconds total (Ref. 4). This delay time includes allowances for Keowee Hydro Unit starting, Emergency Core Cooling Systems (ECCS) pump starts, and valve openings. Similarly, the RB Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system analyzed.

Accident analyses rely on automatic ESPS actuation for protection of the core temperature and containment pressure limits and for limiting off site dose levels following an accident. These include LOCA, and MSLB events that result in RCS inventory reduction or severe loss of RCS cooling.

The ESPS channels satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5).

LCO

The LCO requires three analog channels of ESPS instrumentation for each Parameter in Table 3.3.5-1 to be OPERABLE in each ESPS digital automatic actuation logic channel. Failure of any instrument renders the affected analog channel(s) inoperable and reduces the reliability of the affected Functions.

Only the Allowable Value is specified for each ESPS Function in the LCO. Nominal trip setpoints are specified in the setpoint calculations. The nominal trip setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in

(continued)

BASES

LCO (continued)

the safety analysis to account for instrument uncertainties appropriate to the trip Parameter. These uncertainties are defined in Reference 3.

The Allowable Values for bypass removal functions are stated in the Applicable MODES or Other Specified Condition column of Table 3.3.5-1.

Three ESPS analog instrumentation channels shall be OPERABLE to ensure that a single failure in one analog channel will not result in loss of the ability to automatically actuate the required safety systems.

The bases for the LCO on ESPS Parameters include the following.

Three analog channels of RCS Pressure – Low, RCS Pressure – Low Low, RB Pressure – High and RB Pressure – High High are required OPERABLE. Each analog channel includes a sensor, trip bistable, bypass bistable, bypass relays, and output relays. Failure of a bypass bistable or bypass circuitry, such that an analog channel cannot be bypassed, does not render the analog channel inoperable since the analog channel is still capable of performing its safety function, i.e., this is not a safety related bypass function.

APPLICABILITY

Three analog channels of ESPS instrumentation for each of the following Parameters shall be OPERABLE.

1. <u>Reactor Coolant System Pressure – Low</u>

The RCS Pressure – Low actuation Parameter shall be OPERABLE during operation at or above 1750 psig. This requirement ensures the capability to automatically actuate safety systems and components during conditions indicative of a LOCA or secondary unit overcooling. Below 1750 psig, the low RCS Pressure actuation Parameter can be bypassed to avoid actuation during normal unit cooldowns when safety systems actuations are not required.

(continued)

BASES

APPLICABILITY

1. Reactor Coolant System Pressure - Low (continued)

The allowance for the bypass is consistent with the transition of the unit to a lower energy state, providing greater margins to safety limits. The unit response to any event, given that the reactor is already tripped, will be less severe and allows sufficient time for operator action to provide manual safety system actuations. This is even more appropriate during unit heatups when the primary system and core energy content is low, prior to power operation.

In MODES 5 and 6, there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. RCS pressure and temperature are very low, and many ES components are administratively controlled or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

2. Reactor Coolant System Pressure - Low Low

The RCS Pressure - Low Low actuation Parameter shall be OPERABLE during operation above 900 psig. This requirement ensures the capability to automatically actuate safety systems and components during conditions indicative of a LOCA or secondary unit overcooling. Below 900 psig, the low low RCS Pressure actuation Parameter can be bypassed to avoid actuation during normal unit cooldowns when safety system actuations are not required.

The allowance for the bypass is consistent with the transition of the unit to a lower energy state, providing greater margins to safety limits. The unit response to any event, given that the reactor is already tripped, will be less severe and allows sufficient time for operator action to provide manual safety system actuations. This is even more appropriate during unit heatups when the primary system and core energy content is low, prior to power operation.

(continued)

BASES

BACKGROUND (continued)

The small break LOCA analyses assume a conservative 48 second delay time for the actuation of high pressure injection (HPI) and low pressure injection (LPI) in UFSAR, Chapter 15 (Ref. 2). The large break LOCA analyses assume LPI flow starts in 38 seconds while full LPI flow does not occur until 15 seconds later, or 53 seconds total (Ref. 2). This delay time includes allowances for Keowee Hydro Unit startup and loading, ECCS pump starts, and valve openings. Similarly, the reactor building (RB) Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system.

The ESPS automatic initiation of Engineered Safeguards (ES) Functions to mitigate accident conditions is assumed in the accident analysis and is required to ensure that consequences of analyzed events do not exceed the accident analysis predictions. Automatically actuated features include HPI, LPI, RB Cooling, RB Spray, and RB Isolation.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on automatic ESPS actuation for protection of the core and RB and for limiting off site dose levels following an accident. The digital automatic actuation logic is an integral part of the ESPS.

The ESPS digital automatic actuation logic channels satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The digital automatic actuation logic channels are required to be OPERABLE whenever conditions exist that could require ES protection of the reactor or the RB. This ensures automatic initiation of the ES required to mitigate the consequences of accidents.

The required Function is provided by two associated digital channels as indicated in the following table:

(continued)

BASES

LCO

8. Containment Isolation Valve Position (continued)

Open-Not Open control switch indication via indicating lights in the control room.

9. Containment Area Radiation (High Range)

Containment Area Radiation (High Range) instrumentation is a Type C, Category 1 variable provided to monitor the potential for significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. The Containment Area Radiation instrumentation consists of two channels (RIA 57 and 58) with readout on two indicators and one channel recorded. The indicated range is 1 to 10^7 R/hr.

10. Containment Hydrogen Concentration

Containment Hydrogen Concentration instrumentation is a Type A, Category 1 variable provided to detect high hydrogen concentration conditions that represent a potential for containment breach. This variable is also important in verifying the adequacy of mitigating actions. The Containment Hydrogen Concentration instrumentation consists of two channels (MT 80 and 81) with readout on two indicators and one channel recorded. The indicated range is 0 to 10% hydrogen concentration.

11. Pressurizer Level

Pressurizer Level instrumentation is a Type A, Category 1 variable used in combination with other system parameters to determine whether to terminate safety injection (SI), if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the unit conditions necessary to establish natural circulation in the RCS and to verify that the unit is maintained in a safe shutdown condition. The Pressurizer Level instrumentation consists of three channels (two for Train A and one for Train B) with two channels indicated and one channel recorded.

(continued)

BASES

LCO

16. Core Exit Temperature (continued)

CETs for each of the ICCM trains. Two channels ensure that a single failure will not disable the ability to determine the representative core exit temperature.

17. Subcooling Monitor

The Subcooling Monitor is a Type A, Category 1 variable provided for verification and long term surveillance of core cooling. This variable is a computer calculated value using various inputs from the Primary System.

Two channels of indication are provided. An operable Subcooling Monitor shall consist of: 1) One direct indication from one channel for RCS Loop Saturation margin and one direct indication from the other channel for Core Saturation margin, or 2) One direct indication from each of the two channels for RCS Loop Saturation margin. The indication readouts are located in the control room. This variable also inputs to the unit computer through isolation buffers and is available for trend recording upon operator demand. The range of the readouts is 200°F subcooled to 50°F superheat. The control room display is through the ICCM plasma display unit.

A backup method for determining subcooling margin ensures the capability to accurately monitor RCS subcooling margin (Refer to Specification 5.5.17).

18. HPI System Flow

HPI System Flow instrumentation is a Type A, Category 1 variable provided to support action for short term cooling requirements, to prevent HPI pump runout and inadequate NPSH, and to indicate the need for flow cross connect. HPI flow is throttled based on RCS pressure, subcooled margin, and pressurizer level. Flow measurement is provided by one channel

(continued)

BASES

LCO

18. HPI System Flow (continued)

per train with readout on an indicator and recorder. There are two HPI trains. The channels provide flow indication over a range of 0 to 750 gpm.

19. LPI System Flow

LPI System Flow instrumentation is a Type A, Category 1 variable provided to support action for long term cooling requirements, to prevent LPI pump runout and for flow balance. The indication is also used to identify an LPI pump operating at system pressures above its shutoff head. Flow measurement is provided by one channel per train with readout on an indicator and recorder. There are two LPI trains. The LPI channels provide flow indication over a range of 0 to 6000 gpm.

20. Reactor Building Spray Flow

Reactor Building Spray Flow instrumentation is a Type A, Category 1 variable provided to support action for long term cooling requirements and iodine removal and to prevent Reactor Building Spray and LPI pump runout. Flow measurement is provided by one channel per train with readout on an indicator and recorder. There are two RBS trains. The channels provide flow indication over a range from 0 to 2000 gpm.

21. Emergency Feedwater Flow

EFW Flow instrumentation is a Type D, Category 1 variable provided to monitor operation of RCS heat removal via the SGs. Two channels provide indication of EFW Flow to each SG over a range of approximately 100 gpm to 1200 gpm. Redundant monitoring capability is provided by the two independent channels of instrumentation for each SG. Each flow transmitter provides an input to a control room indicator. One channel also provides input to a recorder.

(continued)

BASES

LCO

21. Emergency Feedwater Flow (continued)

EFW Flow is the primary indication used by the operator to verify that the EFW System is delivering the correct flow to each SG. However, the primary indication used by the operator to ensure an adequate inventory is SG level.

22. Low Pressure Service Water (LPSW) flow to LPI Coolers

LPSW flow to LPI Coolers is a Type A, Category 1 variable is provided to prevent LPSW pump runout and inadequate NPSH. LPSW flow to LPI Coolers is throttled to maintain proper flow balance in the LPSW System.

Flow measurement is provided by one channel per train with readout on an indicator and recorder. The channels provide flow indication over a range from 0-8000 gpm.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required to mitigate accidents and transients. The applicable accidents and transients are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, unit conditions are such that the likelihood of an event occurring that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 is added to the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a unit shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident utilizing alternate instruments and methods, and the low probability of an event requiring these instruments.

(continued)

BASES

ACTIONS (continued)

Note 2 is 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 3.3.8-1. When the Required Channels for a function in Table 3.3.8-1 are specified on a "per" basis (e.g., per loop, per SG, per penetration flow path), then the Condition may be entered separately for each loop, SG, penetration flow path, etc., as appropriate. The Completion Time(s) of the inoperable channels of a Function are tracked separately for each Function starting from the time the Condition is entered for that Function.

<u>A.1</u>

When one or more Functions have one required channel inoperable, the inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience. This takes into account the remaining OPERABLE channel, the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

Condition A is modified by a Note indicating this Condition is not applicable to PAM Functions 14, 18, 19, 20, and 22.

<u>B.1</u>

Required Action B.1 specifies initiation of action described in Specification 5.6.6 that requires a written report to be submitted to the NRC. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability and given the likelihood of unit conditions that would require information provided by this instrumentation. The Completion Time of "Immediately" for Required Action B.1 ensures the requirements of Specification 5.6.6 are initiated.

(continued)

BASES

ACTION (continued)

C.1

When one or more Functions have two required channels inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. This Condition does not apply to the hydrogen monitor channels. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation action operation and the availability of alternative means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance of qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

Condition C is modified by a Note indicating this Condition is not applicable to PAM Functions 10, 14, 18, 19, 20, and 22.

D.1

When two required hydrogen monitor channels are inoperable, Required Action D.1 requires one channel to be restored to OPERABLE status. This action restores the monitoring capability of the hydrogen monitor. The 72 hour Completion Time is based on the relatively low probability of an event requiring hydrogen monitoring. Continuous operation with two required channels inoperable is not acceptable because alternate indications are not available.

Condition D is modified by a Note indicating this Condition is only applicable to PAM Function 10.

E.1

When one required BWST water level channel is inoperable, Required Action E.1 requires the channel to be restored to OPERABLE status. The 24 hour Completion Time is based on the relatively low probability of an event requiring BWST water and the availability of the remaining BWST water level

(continued)

ACTIONS

E.1 (continued)

channel. Continuous operation with one of the two required channels inoperable is not acceptable because alternate indications are not available. This indication is crucial in determining when the water source for ECCS should be swapped from the BWST to the reactor building sump.

Condition E is modified by a Note indicating this Condition is only applicable to PAM Function 14.

F.1

When a flow instrument channel is inoperable, Required Action F.1 requires the affected HPI, LPI, or RBS train to be declared inoperable and the requirements of LCO 3.5.2, LCO 3.5.3, or LCO 3.6.5 apply. For Function 22, LPSW flow to LPI coolers, the affected train is the associated LPI train. The required Completion Time for declaring the train(s) inoperable is immediately. Therefore, LCO 3.5.2, LCO 3.5.3, or LCO 3.6.5 is entered immediately, and the Required Actions in the LCOs apply without delay. This action is necessary since there is no alternate flow indication available and these flow indications are key in ensuring each train is capable of performing its function following an accident. HPI, LPI, and RBS train OPERABILITY assumes that the associated PAM flow instrument is OPERABLE because this indication is used to throttle flow during an accident and assure runout limits are not exceeded or to ensure the associated pumps do not exceed NPSH requirements.

Condition F is modified by a Note indicating this Condition is only applicable to PAM Functions 18, 19, 20, and 22.

G.1

Required Action G.1 directs entry into the appropriate Condition referenced in Table 3.3.8-1. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met the Required Action and associated Completion Time of Condition C, D, or E, as applicable, Condition G is entered for that channel and provides for transfer to the appropriate subsequent Condition.

(continued)

BASES

ACTIONS (continued)

H.1 and H.2

If the Required Action and associated Completion Time of Conditions C, D or E are not met and Table 3.3.8-1 directs entry into Condition H, the unit must be brought to a MODE in which the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and MODE 4 within 18 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.

I.1

If the Required Action and associated Completion Time of Condition C, D or E are not met and Table 3.3.8-1 directs entry into Condition I, alternate means of monitoring the parameter should be applied and the Required Action is not to shut down the unit, but rather to follow the directions of Specification 5.6.6 in the Administrative Controls section of the Technical Specifications. These alternative means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allowed time. The report provided to the NRC should discuss the alternative means used, describe the degree to which the alternative means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

Both the RCS Hot Leg Level and the Reactor Vessel Level are methods of monitoring for inadequate core cooling capability. The subcooled margin monitors (SMM), and core-exit thermocouples (CET) provide an alternate means of monitoring for this purpose. The function of the ICC instrumentation is to increase the ability of the unit operators to diagnose the approach to and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory.

The alternate means of monitoring the Reactor Building Area Radiation (High Range) consist of a combination of installed area radiation monitors and portable instrumentation.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs apply to each PAM instrumentation Function in Table 3.3.8-1 except where indicated.

SR 3.3.8.1

Performance of the CHANNEL CHECK once every 31 days for each required instrumentation channel that is normally energized ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel with 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. CHANNEL CHECK will detect gross channel failure; therefore, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared with similar unit instruments located throughout the unit. If the radiation monitor uses keep alive sources or check sources OPERABLE from the control room, the CHANNEL CHECK should also note the detector's response to these sources.

Agreement criteria are 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. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Offscale low current loop channels are, where practical, verified to be reading at the bottom of the range and not failed downscale.

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 this LCO's required channels.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.2 and SR 3.3.8.3

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. This test verifies the channel responds to measured parameters within the necessary range and accuracy.

Note 1 to SR 3.3.8.3 clarifies that the neutron detectors are not required to be tested as part of the CHANNEL CALIBRATION. There is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices, with minimal drift. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration and the monthly axial channel calibration.

For the Containment Area Radiation instrumentation, a CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr, and a one point calibration check of the detector below 10 R/hr with a gamma source.

Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD)sensors or Core Exit thermocouple sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

SR 3.3.8.2 is modified by a Note indicating that it is applicable only to Functions 7, 10 and 22. SR 3.3.8.3 is modified by Note 2 indicating that it is not applicable to Functions 7, 10 and 22. The Frequency of each SR is based on operating experience and is justified by the assumption of the specified calibration interval in the determination of the magnitude of equipment drift.

REFERENCES

1. Duke Power Company letter from Hal B. Tucker to Harold M. Denton (NRC) dated September 28, 1984.

2. UFSAR, Section 7.5.

(continued)

BASES

REFERENCES (continued)

3. NRC Letter from Helen N. Pastis to H. B. Tucker, "Emergency Response Capability - Conformance to Regulatory Guide 1.97," dated March 15, 1988.

4. Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3, May 1983.

5. NUREG-0737, "Clarification of TMI Action Plan Requirements," 1980.

6. 10 CFR 50.36.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The limiting accident for the EPSL transfer functions is a LOCA with a simultaneous loss of offsite power (Ref. 1). The loss of offsite power is considered to occur coincident with ES actuation. In this scenario, the Load Shed and Transfer to Standby function reenergizes the affected unit's MFBs from the standby buses which are powered from Keowee or Lee. The analyses assume that the maximum time the MFBs will be deenergized is 33 seconds. This time is derived from the 53 second time requirement for full LPI injection minus the 15 second ECCS valve stroke time requirement and 5 seconds for the pump to get to rated speed. EPSL automatic transfer functions are part of the primary success path and function to mitigate an accident or transient that presents a challenge to the integrity of a fission product barrier. The EPSL automatic transfer function satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).
LCO	Two channels of the Automatic Transfer Function, with one channel consisting of Channel A of the Load Shed and Transfer to Standby function and Channel A of the Retransfer to Startup function and the other consisting of Channel B of both of these functions, are required to be OPERABLE. Failure of one channel reduces the reliability of the affected Functions. The requirement for two channels to be OPERABLE ensures that one channel of the function will remain OPERABLE if a single failure has occurred. The remaining channel can perform the safety function.
APPLICABILITY	The automatic transfer function of EPSL is required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that power is provided from AC Sources to the AC Distribution system within the time assumed in the accident analyses. The EPSL automatic transfer function is not required to be OPERABLE in MODES 5 and 6 since more time is available for the operator to respond to a loss of power event.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.22 Emergency Power Switching Logic (EPSL) Manual Keowee Emergency Start Function

BASES

BACKGROUND

The Keowee Emergency Start function of EPSL provides a start signal to the two on-site emergency power sources and sets up controls for the emergency mode. There are two channels of the Emergency Start function. Each channel is capable of starting both Keowee units and activating the controls for the emergency mode.

The Emergency Start channels 1 and 2 are actuated from Engineered Safeguards channels 1 and 2 respectively. The Emergency Start channels can also be activated manually from each control room (i.e., two emergency start switches in the Unit 1 and 2 control room and two emergency start switches in the Unit 3 control room) or cable spread rooms. There are two independent channels associated with each Oconee unit. The two emergency start switches in the ONS Unit 1 and 2 control room initiate through the ONS Unit 1 emergency start channels. Neither start switch provides a signal to the ONS Unit 2 emergency start channels. Therefore, neither can be used to satisfy the LCO requirements for ONS Unit 2. The keylock switches on the emergency start panels in its cable spread room are used to satisfy the ONS Unit LCO requirement.

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the Manual Keowee Emergency Start Function during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The EPSL Manual Keowee Emergency Start Function satisfies Criterion 3 of 10 CFR 50.36 (Ref. 1).

LCO

One channel of the Manual Keowee Emergency Start function, consisting of a manual initiation switch and an Emergency Start channel, is required to be OPERABLE. The emergency

(continued)

BASES

LCO (continued)

start switches in the Unit 1 and 2 Control Room cannot be credited as a manual initiation switch for Unit 2.

APPLICABILITY

The Manual Keowee Emergency Start function required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provides assurance that:

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

b. Systems needed to mitigate a fuel handling accident 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.

ACTIONS

A.1

If the required Manual Keowee Emergency Start channel is inoperable, both Keowee Hydro Units must be declared inoperable immediately. Therefore LCO 3.8.2 is entered immediately, and the required Completion Times for the appropriate Required Actions apply without delay.

SURVEILLANCE REQUIREMENTS

SR 3.3.22.1

A CHANNEL FUNCTIONAL TEST is performed on the required Manual Keowee Emergency Start channel to ensure the channel will perform its function. The Frequency of 12 months is based on engineering judgment and operating experience that determined testing on a 12 month interval provides reasonable assurance that the circuitry is available to perform its safety function.

(continued)

BASES (continued)

REFERENCES	1.	10 CFR 50.36.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.5

A CHANNEL FUNCTIONAL TEST is required within 12 hours after decreasing RCS temperature to ≤ 325°F and every 31 days thereafter to ensure the setpoint is proper for using the PORV for LTOP. PORV actuation is not needed, as it could depressurize the RCS.

The 12 hour Frequency considers the unlikelihood of a low temperature overpressure event during the time. The 31 day Frequency is based on industry accepted practice and is acceptable by experience with equipment reliability.

SR 3.4.12.6

Verification that administrative controls, other than limits for pressurizer level, that assure ≥ 10 minutes are available for operator action to mitigate the consequences of an LTOP event are implemented is necessary every 12 hours. This verification consists of a combination of administrative checks for alarm availability, appropriate restrictions on pressurizer level, controls for High Pressure Nitrogen, etc., as well as visual confirmation using available indications that associated physical parameters are within limits.

The Frequency is shown by operating practice sufficient to regularly assess indications of potential degradation and verify operation within the safety analysis.

SR 3.4.12.7

The performance of a CHANNEL CALIBRATION is required every 18 months. The CHANNEL CALIBRATION for the LTOP setpoint ensures that the PORV will be actuated at the appropriate RCS pressure by verifying the accuracy of the instrument string. The calibration can only be performed in shutdown.

The Frequency considers a typical refueling cycle and industry accepted practice.

(continued)

BASES

BACKGROUND (continued)

The CFTs thus form a passive system for injection directly into the reactor vessel. Except for the core flood line break LOCA, a unique accident that also disables a portion of the injection system, both tanks are assumed to operate in the safety analyses for Design Basis Events. Because injection is directly into the reactor vessel downcomer, and because it is a passive system not subject to the single active failure criterion, all fluid injection is credited for core cooling.

The CFT gas/water volumes, gas pressure, and outlet pipe size are selected to provide core cooling for a large break LOCA prior to the injection of coolant by the LPI System.

APPLICABLE SAFETY ANALYSES

The CFTs are credited in both the large and small break LOCA analyses (Ref. 1). These accident analyses establish the acceptance limits for the CFTs. In performing the LOCA calculations, conservative assumptions are made concerning the availability of emergency injection flow. The assumption of the loss of offsite power is required by regulations. In the early stages of a LOCA with the loss of offsite power, the CFTs provide the sole source of makeup water to the RCS.

This is because the LPI pumps and high pressure injection (HPI) pumps cannot deliver rated flow until the Keowee Hydro Units start and come to rated speed and valves open.

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the reactor coolant pump. During this event, the CFTs discharge to the RCS as soon as RCS pressure decreases below CFT pressure. As a conservative estimate, no credit is taken for HPI for large break LOCAs. LPI is not assumed to occur until 38 seconds after loss of offsite power occurs with full LPI flow not occurring until 15 seconds later, or 53 seconds total. No operator action is assumed during the blowdown stage of a large break LOCA.

The small break LOCA analysis also assumes a time delay after Engineered Safeguards actuation before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated by the CFTs, with pumped flow then

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

providing continued cooling. As break size decreases, the CFTs and HPI pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the CFTs continues to decrease until the tanks are not required and the HPI pumps become responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature of 2200°F;

b. Maximum cladding oxidation of ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction of ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core maintained in a coolable geometry.

Since the CFTs discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

The limits for operation with a CFT that is inoperable for any reason other than the boron concentration not being within limits minimize the time that the unit is exposed to a LOCA event occurring along with failure of a CFT, which might result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be opened, or the proper water volume or nitrogen cover pressure cannot be restored, the full capability of one CFT is not available and prompt action is required to place the reactor in a MODE in which this capability is not required.

In addition to LOCA analyses, the CFTs have been assumed to operate to provide borated water for reactivity control for severe overcooling events such as a main steam line break (MSLB).

(continued)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 High Pressure Injection (HPI)

BASES

BACKGROUND

The function of the ECCS is to provide core cooling to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA);

b. Rod ejection accident (REA);

c. Steam generator tube rupture (SGTR); and

d. Main steam line break (MSLB).

There are two phases of ECCS operation: injection and recirculation. In the injection phase, all injection is initially added to the Reactor Coolant System (RCS) via the cold legs or Core Flood Tank (CFT) lines to the reactor vessel. After the borated water storage tank (BWST) has been depleted, the recirculation phase is entered as the suction is transferred to the reactor building sump.

The HPI System consists of two independent trains, each of which splits to discharge into two RCS cold legs, so that there are a total of four HPI injection lines. Each train takes suction from the BWST, and has an automatic suction valve and discharge valve which open upon receipt of an Engineered Safeguards Protective System (ESPS) signal. The two HPI trains are designed and aligned such that they are not both susceptible to any single active failure including the failure on any power operating component to operate or any single failure of electrical equipment. There are three ESPS actuated HPI pumps, each of which can provide flow to either train. At least one pump is normally running providing RCS makeup and seal injection to the reactor coolant pumps. Suction header cross-connect valves are normally open, and discharge header cross-connect valves are normally closed. Additional discharge valves (HPI discharge crossover valves) can be used to bypass the normal discharge valves and assure the ability to feed either train's injection lines from the pump(s) on the other train. A safety grade flow indicator is provided for the flow path associated with each of these four discharge valves. These

(continued)

BASES

LCO (continued)

may be called upon to mitigate the consequences of other transients and accidents. Each HPI train includes the piping, instruments, pumps, valves, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an ESPS signal. The safety grade flow indicator associated with the normal discharge valve is required to be OPERABLE to support the associated HPI train's automatic OPERABILITY. Each LPI-HPI flow path includes the piping, instruments, pumps, valves and controls to ensure the capability to manually transfer suction to the reactor building sump (LPI-HPI flow path).

During an event requiring HPI actuation, a flow path is provided to ensure an abundant supply of water from the BWST to the RCS via the HPI pumps and their respective discharge flow paths to each of the four cold leg injection nozzles and the reactor vessel. In the long term, this flow path may be manually transferred to take its supply from the reactor building sump and to supply borated water to the RCS via the LPI-HPI flow path (piggy-back mode).

The flow path for each HPI train must maintain its designed independence to ensure that no single active failure can disable both HPI trains.

The LCO is modified by a Note that requires three HPI pumps and the HPI discharge crossover valves (HP-409 and HP-410) to be OPERABLE and the suction header to be cross-connected when THERMAL POWER is > 60% RTP. The safety grade flow indicator associated with a HPI discharge crossover valve is required to be OPERABLE to support HPI discharge crossover valve OPERABILITY. The Note modifies the pump and valve OPERABILITY and valve alignment requirements to provide additional requirements assumed by the safety analyses at power levels > 60% RTP.

APPLICABILITY

In MODES 1 and 2, and MODE 3 with RCS temperature > 350°F, the HPI train OPERABILITY requirements for the small break LOCA are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The HPI pump performance is based on the small break LOCA, which establishes the pump

(continued)

BASES

BACKGROUND (continued)

The LPI pumps are capable of discharging to the RCS at an RCS pressure of approximately 200 psia. When the BWST has been nearly emptied, the suction for the LPI pumps is manually transferred to the reactor building sump.

In the long term cooling period, flow paths in the LPI System are established to preclude the possibility of boric acid in the core region reaching an unacceptably high concentration. Two gravity flow paths are available by means of a drain line from the hot leg to the Reactor Building sump which draws coolant from the top of the core, thereby inducing core circulation. The system is designed with redundant drain lines.

During a large break LOCA, RCS pressure will rapidly decrease. The LPI System is actuated upon receipt of an ESPS signal. If offsite power is available, the safeguard loads start immediately. If offsite power is not available, the Engineered Safeguards (ES) buses are connected to the Keowee Hydro Units. The time delay (38 seconds) associated with Keowee Hydro Unit startup and pump starting determines the time required before pumped flow is available to the core following a LOCA. Full LPI flow is not available until the LPI valve strokes full open.

The LPI and HPI (LCO 3.5.2, "High Pressure Injection (HPI)"), along with the passive CFTs and the BWST covered in LCO 3.5.1, "Core Flood Tanks (CFTs)," and LCO 3.5.4, "Borated Water Storage Tank (BWST)," provide the cooling water necessary to meet 10 CFR 50.46 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 1), will be met following a LOCA:

a. Maximum fuel element cladding temperature is $\leq 2200°F$;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also helps ensure that reactor building temperature limits are met.

The LPI System is assumed to provide injection in the large break LOCA analysis at full power (Ref. 2). This analysis establishes a minimum required flow for the LPI pumps, as well as the minimum required response time for their actuation.

The large break LOCA event assumes a loss of offsite power and a single failure (loss of the CT-4 transformer). For analysis purposes, the loss of offsite power assumption may be conservatively inconsistent with the assumed operation of some equipment, such as reactor coolant pumps (Ref. 3). During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the reactor building. The nuclear reaction is terminated by moderator voiding during large breaks. Following depressurization, emergency cooling water is injected into the reactor vessel core flood nozzles, then flows into the downcomer, fills the lower plenum, and refloods the core.

In the event of a Core Flood line break which results in a LOCA, with a concurrent single failure on the unaffected LPI train opposite the Core Flood break, the LPI discharge header crossover valves (LP-9 and LP-10) must be capable of being manually opened. The LPI cooler outlet throttle valves and LPI header isolation valves must be capable of being manually opened to provide assurance that flow can be established in a timely manner even if the capability to operate them from the control room is lost. These manual actions will allow cross-connection of the LPI pump discharge to the intact LPI/Core Flood tank header to provide abundant emergency core cooling.

The safety analyses show that an LPI train will deliver sufficient water to match decay heat boiloff rates for a large break LOCA.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

In the large break LOCA analyses, full LPI is not credited until 53 seconds after actuation of the ESPS signal. This is based on a loss of offsite power and the associated time delays in Keowee Hydro Unit startup, valve opening and pump start. Further, LPI flow is not credited until RCS pressure drops below the pump's shutoff head. For a large break LOCA, HPI is not credited at all.

The LPI trains satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4).

LCO

In MODES 1, 2, and 3, two independent (and redundant) LPI trains are required to ensure that at least one LPI train is available, assuming a single failure in the other train. Additionally, individual components within the LPI trains may be called upon to mitigate the consequences of other transients and accidents. Each LPI train includes the piping, instruments, pumps, valves, heat exchangers and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an ES signal and the capability to manually (remotely) transfer suction to the reactor building sump. The safety grade flow indicator associated with an LPI train is required to be OPERABLE to support LPI train OPERABILITY. The safety grade flow indicator associated with LPSW flow to an LPI cooler is required to be OPERABLE to support LPI train OPERABILITY.

In MODE 4, one of the two LPI trains is required to ensure sufficient LPI flow is available to the core.

During an event requiring LPI injection, a flow path is required to provide an abundant supply of water from the BWST to the RCS, via the LPI pumps and their respective supply headers, to the reactor vessel. In the long term, this flow path may be switched to take its supply from the reactor building sump.

This LCO is modified by three Notes. Note 1 changes the LCO requirement when in MODE 4 for the number of OPERABLE trains from two to one. Note 2 allows an LPI train to be considered OPERABLE during alignment, when aligned or when operating for decay heat removal if capable of being manually (remotely) realigned to the LPI mode of operation. This provision is necessary because of the dual requirements

(continued)

BASES

LCO (continued)

of the components that comprise the LPI and decay heat removal modes of the LPI System. Note 3 requires the LPI discharge header crossover valves (LP-9 and LP-10) to be OPERABLE in MODES 1, 2, and 3.

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both LPI trains. If both LPI discharge header crossover valves (LP-9 and LP-10) are simultaneously open then only one LPI train is considered OPERABLE.

APPLICABILITY

In MODES 1, 2 and 3, the LPI train OPERABILITY requirements for the Design Basis Accident, a large break LOCA, are based on full power operation. The LPI discharge crossover valve OPERABILITY requirements for CFT line break is based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES.

In MODE 4, one OPERABLE LPI train is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

In MODES 5 and 6, unit conditions are such that the probability of an event requiring LPI injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "DHR and Coolant Circulation - High Water Level," and LCO 3.9.5, "DHR and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With one LPI train inoperable in MODES 1, 2 or 3, the inoperable train must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on NRC recommendations (Ref. 5) that are based on a risk evaluation and is a reasonable time for many repairs. This reliability analysis has shown the risk of having one LPI train

(continued)

BASES

ACTIONS

A.1 (continued)

inoperable to be sufficiently low to justify continued operation for 72 hours.

B.1

With one or more LPI discharge crossover valves inoperable, the inoperable valve(s) must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on NRC recommendations (Ref. 5) that are based on a risk evaluation and is a reasonable time for many repairs.

C.1

If the Required Action and associated Completion Time of Condition A or B are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and MODE 4 within 60 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

With one required LPI train inoperable in MODE 4, the unit is not prepared to respond to an event requiring low pressure injection and may not be prepared to continue cooldown using the LPI pumps and LPI heat exchangers. The Completion Time of immediately, which would initiate action to restore at least one LPI train to OPERABLE status, ensures that prompt action is taken to restore the required LPI capacity. Normally, in MODE 4, reactor decay heat must be removed by a decay heat removal (DHR) loop operating with suction from the RCS. If no LPI train is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generator(s).

(continued)

BASES

BACKGROUND (continued)

2. closed by manual valves, blind flanges, or de-activated automatic valves in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks"; and

c. The equipment hatch is closed.

APPLICABLE SAFETY ANALYSES

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

The accidents that result in a challenge to containment from high pressures and temperatures are a loss of coolant accident (LOCA) and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA. In the accident analyses, it is assumed that the containment is OPERABLE such that, for the accidents involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option A and B (Ref. 1), as L_a: the maximum allowable leakage rate at the calculated maximum peak containment pressure (P_a) resulting from the limiting accident. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.25% per day in the safety analysis at P_a = 59.0 psig (Ref. 3).

The containment satisfies Criterion 3 of the 10 CFR 50.36 (Ref. 4).

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and Type A leakage rate test requirements of the Containment Leakage Rate Testing Program. As left leakage prior to the first startup after performing a required leakage test is required to be $< 0.75\ L_a$ for overall Type A leakage following an outage or shutdown that included Type A testing. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

Maintaining the containment OPERABLE requires compliance with the Type B and C leakage rate test requirements of 10 CFR 50, Appendix J, Option A (Ref. 1), as modified by approved exemptions. As left leakage prior to the first startup after performing a required 10 CFR 50, Appendix J, Option A, leakage test is required to be $< 0.6\ L_a$ for combined Type B and C leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by Appendix J, Option A, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.3

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are as described in Specification 5.5.7, "Pre-stressed Containment Tendon Surveillance Program."

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The accident that results in a release of radioactive material within containment is a loss of coolant accident (LOCA) (Ref. 2). In the analysis of this accident, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option A and B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a) following an accident. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4).

LCO

Each containment air lock forms part of the containment pressure boundary. As a part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from an accident. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are normally closed when the air lock is not being used for normal entry into or exit from containment.

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J, Option A (Ref. 1), as modified by approved exemptions. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by 10 CFR 50, Appendix J, Option A, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply. Either a full air lock leak test or a leak test of the outer air lock door seal performed within 3 days of initial opening, and during periods of frequent use, at least once every 3 days, is an acceptable method of complying with 10 CFR 50, Appendix J requirements (References 5 and 6).

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable, since either air lock door is capable of providing a fission product barrier in the event of an accident. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.2. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2 (continued)

containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry or exit (procedures require strict adherence to single door opening), this test is only required to be performed every 18 months. 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 loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. The 18 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during use of the air lock.

REFERENCES

1. 10 CFR 50, Appendix J, Option A and B.
2. UFSAR, Section 15.14.
3. UFSAR, Section 6.2.
4. 10 CFR 50.36.
5. Duke Power Company letter from William O. Parker, Jr. to Harold R. Denton (NRC) dated July 24, 1981.
6. NRC Letter from Philip C. Wagner to William O. Parker, Jr., dated November 6, 1981, Issuance of Amendment 104, 104 and 101 to Licenses DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station Units Nos 1, 2 and 3.

BASES

BACKGROUND (continued)

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analysis. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analysis will be maintained.

The Reactor Building Purge System is part of the Reactor Building Ventilation System. The Purge System was designed for intermittent operation, providing a means of removing airborne radioactivity caused by minor leakage from the RCS prior to personnel entry into containment. The Reactor Building Purge System consists of one 48 inch line for exhaust and one 48 inch line for supply, with exhaust fans capable of purging the containment atmosphere at a rate of approximately 35,000 ft^3/min. The reactor building purge supply and exhaust lines each contain two isolation valves that receive a reactor building isolation signal.

Failure of the purge valves to close following a design basis event would cause a significant increase in the radioactive release because of the large containment leakage path introduced by these 48 inch purge lines. Failure of the purge valves to close would result in leakage considerably in excess of the containment design leakage rate of 0.25% of containment air weight per day (L_a) (Ref. 1). Because of their large size, the 48 inch purge valves are not qualified for automatic closure from their open position under accident conditions. Therefore, the 48 inch purge valves are maintained sealed closed (SR 3.6.3.1) in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analysis of any event requiring isolation of containment is applicable to this LCO.

The accident that results in a significant release of radioactive material within containment is a loss of coolant

(continued)

BASES

LCO (continued)

This LCO provides assurance that the containment isolation valves and purge valves will perform their designated 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, an accident 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, except for 48 inch purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated individual, 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. Due to the size of the reactor building purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow paths containing these valves may not be opened under administrative controls.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

(continued)

BASES (continued)

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, containment pressure must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If the Required Action and associated Completion Time is not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to MODE 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 3.6.4.1

Verifying that containment pressure is within limits ensures that operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed after taking into consideration operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. UFSAR, Chapter 15.
2. UFSAR, Section 6.2.
3. 10 CFR 50, Appendix K.
4. 10 CFR 50.36.

BASES

LCO (continued)

requirements are met, two reactor building spray trains and three reactor building cooling trains must be OPERABLE in MODES 1 and 2. In MODES 3 or 4, one reactor building spray train and two reactor building cooling trains are required to be OPERABLE. The LCO is provided with a note that clarifies this requirement. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst-case single active failure occurs.

Each reactor building spray train shall include a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST (via the LPI System) upon an Engineered Safeguards Protective System signal and manually transferring suction to the reactor building sump. The safety grade flow indicator of an RBS train and the safety grade flow indicator of the associated LPI train are both required to be OPERABLE to support RBS train OPERABILITY.

Each reactor building cooling train shall include cooling coils, fusible dropout plates, an axial vane flow fan, instruments, valves, and controls to ensure an OPERABLE flow path. Valve LPSW-108 shall be locked open to support system OPERABILITY.

APPLICABILITY

In MODES 1, 2, 3, and 4, an accident could cause a release of radioactive material to containment and an increase in containment pressure and temperature, requiring the operation of the reactor building spray trains and reactor building cooling trains.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Reactor Building Spray System and the Reactor Building Cooling System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

<u>A.1</u>

With one reactor building spray train inoperable in MODE 1 or 2, the inoperable reactor building spray train must be restored to OPERABLE status within 7 days. In this

(continued)

BASES

ACTIONS

A.1 (continued)

Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 7 day Completion Time takes into account the redundant heat removal capability afforded by the OPERABLE reactor building spray train, reasonable time for repairs, and the low probability of an accident occurring during this period.

The 14 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, Completion Times, for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

B.1

With one of the reactor building cooling trains inoperable in MODE 1 or 2, the inoperable reactor building cooling train must be restored to OPERABLE status within 7 days. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Reactor Building Spray System and Reactor Building Cooling System and the low probability of an accident occurring during this period.

The 14 day portion of the Completion Time for Required Action B.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.5 Emergency Feedwater (EFW) System

BASES

BACKGROUND

The EFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply. The EFW pumps take suction through suction lines from the upper surge tank (UST) and condenser Hotwell and pump to the steam generator secondary side through the EFW nozzles. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam relief valves (MSRVs) (LCO 3.7.1, "Main Steam Relief Valves (MSRVs)"), or atmospheric dump valves (ADVs). If the main condenser is available, steam may be released via the Turbine Bypass System and recirculated to the condenser Hotwell.

The EFW System consists of two motor driven EFW pumps and one turbine driven EFW pump, any one of which can provide the required heat removal capability. Thus, the requirements for diversity in motive power sources for the EFW System are met. The steam turbine driven EFW pump receives steam from either of the two main steam headers, upstream of the main turbine stop valves (TSVs), or from the Auxiliary Steam System which can be supplied from the other two unit's Main Steam System. The EFW System supplies a common header capable of feeding either or both steam generators. The EFW System normally receives a supply of water from the UST. The EFW System can also be aligned to the condenser Hotwell. An additional source of water is the condensate storage tank which can be pumped to the USTs.

The EFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

The three emergency feedwater pumps are started automatically upon a loss of both main feedwater pumps or a signal from the ATWS Mitigation System Actuation Circuitry (AMSAC). The two motor driven emergency feedwater pumps are also started automatically upon a low steam generator level which exists for at least 30 seconds.

(continued)

BASES

BACKGROUND (continued)

The EFW System is discussed in the UFSAR, Section 10.4.7, (Ref. 1).

APPLICABLE SAFETY ANALYSES

The EFW System mitigates the consequences of any event with a loss of normal feedwater.

The design basis of the EFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at 1064 psia for the MDEFW pump and 1100 psig for the TDEFW pump.

The limiting event for the EFW System is the loss of main feedwater with offsite power available.

The EFW System design is such that it can perform its function following a loss of the turbine driven main feedwater pumps combined with a loss of normal or emergency electric power.

The EFW System satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

This LCO provides assurance that the EFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Three independent EFW pumps and two flow paths are required to be OPERABLE to ensure the availability of residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering one pump by a steam driven turbine supplied with steam from a source not isolated by the closure of the TSVs, and two pumps from a power source that, in the event of loss of offsite power, is supplied by the emergency power source.

The EFW System is considered to be OPERABLE when the components and flow paths required to provide EFW flow to the steam generators are OPERABLE. This requires that the turbine driven EFW pump be OPERABLE with a steam supply from either one of the main steam lines upstream of the TSVs or from the Auxiliary Steam System. The two motor driven EFW

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.4 (continued)

This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

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

SR 3.7.8.5

Verification that ESV float valves open upon an actual or simulated actuation ensures a flow path is provided to the ESV pumps to assure the ECCW siphon headers are maintained sufficiently primed. The basis for the Frequency of 92 days is ASME Code, Section XI (Ref. 4).

SR 3.7.8.6

Verification that required ESV valves actuate to the correct position ensures the ESV tank minimum flow valves will automatically close during a loss of offsite power event so that the full capacity of the ESV pumps will be aligned to the ECCW siphon headers. Verification that required SSW valves actuate to the correct position ensures sufficient seal water is provided to ESV pumps. The basis for the Frequency of 92 days is ASME Code, Section XI (Ref. 4).

SR 3.7.8.7

Verifying that each ESV pump's capacity at the test point is greater than or equal to the required capacity ensures that pump performance has not degraded below the acceptance criteria during the cycle. ESV pump capacity is determined by measuring the "apparent" flow rate and calculating the "corrected" flow rate by adjusting for air density changes between the measurement point and the pump inlet. The vacuum level must be within a prescribed range during this measurement to ensure that the flowmeter is on-scale and the pump operating liquid is not cavitating. Note that the pump is a constant volume machine. Thus, there is not a single

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.8 (continued)

test point but a range of acceptable vacuum levels. Although ASME code for inservice testing does not specifically address vacuum pumps, manufacturers test methods coupled with the ASME standard (OM-6) (Ref. 5) requirements for testing methodology are used as a guide for testing. Accordingly, the basis for the Frequency of 92 days is ASME Code, Section XI (Ref. 4).

Verification that each required ESV pump automatically starts within 1200 seconds after an actual or simulated restoration of emergency power assures required ESV pumps will function after a loss of offsite power to maintain ECCW siphon headers sufficiently primed to maintain necessary flow to the suction of LPSW pumps. The Frequency of 18 months is based on engineering judgement.

SR 3.7.8.9

This SR verifies the ECCW system functions to supply siphon header flow to the suction of the LPSW pumps during design basis conditions by ensuring air accumulation in the ECCW siphon headers is within the removal capabilities of the ESV System. For Unit 3 the LPSW System shall be taking its suction from the siphon header during the SR. This SR establishes siphon flow with the ESV pumps off. Air accumulation in the pipe results in a corresponding reduction in water level in the CCW piping over a time period. The rate of water level reduction is recorded and compared to limits established in design basis documents. The limits on the rate of water level reduction over a time period are established to ensure ECCW siphon header air accumulation rate is within the removal capabilities of the ESV System under design basis conditions. The Frequency of 18 months is based on the need to perform this SR when the Unit is shutdown.

A Note states that for Units 1 and 2, the SR is not required to be performed with the shared LPSW System for Units 1 and 2 taking suction from the siphon. This is necessary to avoid potential effects on an operating unit and is acceptable since the capability of the LPSW pumps to take suction from the CCW crossover header is demonstrated by normal, day-to-day operation of the LPSW pumps. Although a

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.9 (continued)

loss of suction to the LPSW pumps is unlikely during this SR, it is prudent to minimize the potential for jeopardizing the LPSW suction supply to the LPSW pumps when they are supporting an operating Unit.

REFERENCES

1. UFSAR, Chapter 9.
2. 10 CFR 50.36.
3. UFSAR, Chapter 16.
4. ASME, Boiler and Pressure Vessel Code, Section XI.
5. ASME Standard OM-6.

BASES

LCO (continued)

An SFPVS train is considered OPERABLE when its associated:

1. Fan is OPERABLE;
2. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and
3. Ductwork and dampers are OPERABLE, and air flow can be maintained.

The LCO is modified by two Notes. Note 1 states LCO 3.0.3 does not apply. If moving fuel or conducting crane operations with load over the storage pool while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel or conducting crane operations with load over the storage pool while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown. Note 2 states the requirements of this LCO is not applicable during reracking operations with no fuel in the spent fuel pool. With no fuel in the spent fuel pool, the potential release of radioactive material to the environs resulting from crane operations with load over the storage pool is substantially reduced.

APPLICABILITY

During movement of fuel in the fuel handling area or during crane operations with loads over the spent fuel pool, the SFPVS is always required to be OPERABLE.

ACTIONS

<u>A.1 and A.2</u>

With one SFPVS train inoperable, the OPERABLE SFPVS train must be started immediately with its discharge through the associated reactor building purge filter or fuel movement in the spent fuel pool and crane operations with loads over the spent fuel pool suspended. This action ensures that the remaining train is OPERABLE, and that any active failures will be readily detected.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

If the system is not placed in operation, this action requires suspension of fuel movement and suspension of crane operation with loads over the spent fuel pool, which precludes a fuel handling accident. This action does not preclude the movement of fuel assemblies or crane loads to a safe position.

B.1

When two trains of the SFPVS are inoperable during movement of fuel in the spent fuel pool, the unit must be placed in a condition in which the LCO does not apply. This Action involves immediately suspending movement of fuel assemblies in the spent fuel pool and suspension of crane operations with loads over the spent fuel pool. This does not preclude the movement of fuel or crane loads to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Systems without heaters need only be operated through the associated reactor building purge filters at a design flow ± 10% for ≥ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy.

SR 3.7.17.2

This SR verifies that the required SFPVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated

(continued)

BASES

LCO (continued)

underground power path are closed to provide single failure protection for the KHUs.

Operable offsite sources are required to be "physically independent" (separate towers) prior to entering the 230 kV switchyard. Once the 230 kV lines enter the switchyard, an electrical pathway must exist through OPERABLE power circuit breakers (PCBs) and disconnects such that both sources are available to energize the Unit's startup transformer either automatically or with operator action. Once within the boundary of the switchyard, the electrical pathway may be the same for both independent offsite sources. In addition, at least one E breaker must be available to automatically supply power to a main feeder bus from the energized startup transformer. The voltage provided to the startup transformer by the two independent offsite sources must be sufficient to ensure ES equipment will operate. Two of the following offsite sources are required:

1) Jocassee (from Jocassee) Black or White,
2) Dacus (from North Greenville) Black or White,
3) Oconee (from Central) Black or White,
4) Calhoun (from Central) Black or White,
5) Autobank transformer fed from either the Asbury (from Newport), Norcross (from Georgia Power), or Katoma (from Jocassee) 525 kV line.

An OPERABLE KHU and its required emergency power path are required to be able to provide sufficient power within specified limits of voltage and frequency within 23 seconds after an emergency start initiate signal and includes its required emergency power path, required instrumentation, controls, auxiliary and DC power, cooling and seal water, lubrication and other auxiliary equipment necessary to perform its safety function. Two emergency power paths are available. One emergency power path consists of an underground circuit while the other emergency power pathway uses an overhead circuit through the 230 kV switchyard.

(continued)

BASES

ACTIONS

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

LCT (using the 100 kV transmission circuit electrically separated from the grid and offsite loads) must energize a standby bus prior to the outage exceeding 24 hours. This ensures the availability of a power source on the standby buses when the KHU and its required underground emergency power path are out of service in excess of 24 hours. The second Completion Time of Required Action D.2 permits the standby buses to be re-energized by an LCT within 1 hour in the event this source is subsequently lost.

The second Completion Time for Required Action D.3 establishes a limit on the maximum time allowed for a KHU to be inoperable during any single contiguous occurrence of having a KHU inoperable. If Condition D is entered as a result of switching an inoperable KHU from the overhead to the underground emergency power path, it may have been inoperable for up to 72 hours. This could lead to a total of 144 hours since the initial failure of the KHU. The second Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time the KHU become inoperable, instead of at the time Condition D was entered.

E.1 and E.2

If the Required Action and associated Completion Time for Required Action D.2 are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours for one Oconee unit and 24 hours for other Oconee unit(s) and to MODE 5 within 84 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 plant systems.

F.1 and F.2

With the zone overlap protection circuitry inoperable when the overhead electrical disconnects for the KHU associated with the underground power path are closed, the zone overlap protection circuitry must be restored to OPERABLE status or

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.2

This SR verifies adequate battery voltage when the KHU batteries are on float charge. This SR is performed to verify KHU battery OPERABILITY. The Frequency of once per 7 days is consistent with manufacturers recommendations and IEEE-450 (Ref. 8).

SR 3.8.1.3

This SR verifies the availability of the KHU associated with the underground emergency power path to start automatically and energize the underground power path. Utilization of either the auto-start or emergency start sequence assures the control function OPERABILITY by verifying proper speed control and voltage. Power path verification is included to demonstrate breaker OPERABILITY from the KHU onto the standby buses. This is accomplished by closing the Keowee Feeder Breakers (SK) to energize each deenergized standby bus. The 31 day Frequency is adequate based on operating experience to provide reliability verification without excessive equipment cycling for testing.

SR 3.8.1.4

This surveillance verifies the availability of the KHU associated with the overhead emergency power path. Utilization of either the auto-start or emergency start sequence assures the control function OPERABILITY by verifying proper speed control and voltage. The ability to supply the overhead emergency power path is satisfied by demonstrating the ability to synchronize (automatically or manually) the KHU with the grid system. The SR also requires that the underground power path be energized after removing the KHU from the overhead emergency power path. This surveillance can be satisfied by first demonstrating the ability of the KHU associated with the underground emergency path to energize the underground path then synchronizing the KHU to the overhead emergency power path. The SR is modified by a Note indicating that the requirement to energize the underground emergency power path is not

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.4 (continued)

applicable when the overhead disconnects are open for the KHU associated with the underground emergency power path or 2) when complying with Required Action D.1. The latter exception is necessary since Required Action D.1 continues to be applicable when both KHUs are inoperable.

The 31 day Frequency for this Surveillance was determined to be adequate based on operating experience to provide reliability verification without excessive equipment cycling for testing.

SR 3.8.1.5

This surveillance verifies OPERABILITY of the trip functions of each closed SL and each closed N breaker. Neither of these breakers have any automatic close functions; therefore, only the trip coils require verification. Cycling of each breaker demonstrates functional OPERABILITY and the coil monitor circuits verify the integrity of each trip coil. The 31 day frequency is based on operating experience.

This SR modified by a Note that states it is not required to be performed for an SL breaker when its standby bus is energized from a LCT via an isolated power path. This is necessary since the standby buses are required to be energized from a LCT by several Required Actions of Specification 3.8.1 and the breakers must remain closed to energized the standby buses from a LCT.

SR 3.8.1.6

Infrequently used source breakers are cycled to ensure OPERABILITY. The Standby breakers are to be cycled one breaker at a time to prevent inadvertent interconnection of two units through the standby bus breakers. Cycling the startup breakers verifies OPERABILITY of the breakers and associated interlock circuitry between the normal and startup breakers. This circuitry provides an automatic, smooth, and safe transfer of auxiliaries in both directions between sources. The 31 day Frequency for this Surveillance was determined to be adequate based on operating experience

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.6 (continued)

to provide reliability verification without excessive equipment cycling for testing.

This SR is modified by a Note which states the SR is not required to be performed for an S breaker when its standby bus is energized from a LCT via an isolated power path. Thisis necessary since the standby buses are required to be energized from a LCT by several Required Actions of Specification 3.8.1 and cycling the S breakers connects the standby buses with the main feeder buses which are energized from another source.

SR 3.8.1.7

The KHU tie breakers to the underground path, ACB3 and ACB4, are interlocked to prevent cross-connection of the KHU generators. The safety analysis utilizes two independent power paths for accommodating single failures in applicable accidents. Connection of both generators to the underground path compromises the redundancy of the emergency power paths. Installed test logic is used to verify a circuit to the close coil on one underground ACB does not exist with the other underground ACB closed. The 12 month Frequency for this surveillance is adequate based on operating experience to provide reliability verification without excessive equipment cycling for testing.

SR 3.8.1.8

Each KHU tie breaker to the underground emergency power path and tie breaker to the overhead emergency path, are interlocked to prevent the unit associated with the underground circuit from automatically connecting to the overhead emergency power path. The safety analysis utilizes two independent power paths for accommodating single failures in applicable accidents. Connection of both generators to the overhead emergency power path compromises the redundancy of the emergency power paths. Temporary test instrumentation is used to verify a circuit to the close coil on the overhead ACB does not exist with the Underground ACB closed. The 12 month Frequency for this Surveillance

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.8 (continued)

was determined to be adequate based on operating experience to provide reliability verification without excessive equipment cycling for testing.

SR 3.8.1.9

This surveillance verifies the KHUs' response time to an Emergency Start signal (normally performed using a pushbutton in the control room) to ensure ES equipment will have adequate power for accident mitigation. UFSAR Section 6.3.3.3 (Ref. 9) establishes the 23 second time requirement for each KHU to achieve rated frequency and voltage. Since the only available loads of adequate magnitude for simulating a accident is the grid, subsequent loading on the grid is required to verify the KHU's ability to assume rapid loading under accident conditions. Sequential block loads are not available to fully test this feature. This is the reason for the requirement to load the KHUs at the maximum practical rate. The 12 month Frequency for this SR is adequate based on operating experience to provide reliability verification without excessive equipment cycling for testing.

SR 3.8.1.10

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements as specified in Reference 4.

The Surveillance Frequency of 12 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 6) and Regulatory Guide 1.129 (Ref. 7), which state that the battery service test should be performed with intervals between tests not to exceed 18 months.

SR 3.8.1.11

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

abnormal deterioration that could potentially degrade battery performance. The 12 month Frequency for this SR is consistent with manufacturers recommendations and IEEE-450 (Ref. 8), which recommends detailed visual inspection of cell condition and rack integrity on a yearly basis.

SR 3.8.1.12

Verification of cell to cell connection cleanliness, tightness, and proper coating with anti-corrosion grease provides an indication of any abnormal condition, and assures continued OPERABILITY of the battery. The 12 month frequency is based on engineering judgement and operational experience and is sufficient to detect cell connection degradation when it is properly coupled with other surveillances more frequently performed to detect abnormalities.

SR 3.8.1.13

The KHU underground ACBs have a control feature which will automatically close the KHU, that is pre-selected to the overhead path, into the underground path upon an electrical fault in the zone overlap region of the protective relaying. This circuitry prevents an electrical fault in the zone overlap region of the protective relaying from locking out both emergency power paths during dual KHU grid generation. In order to ensure this circuitry is OPERABLE, an electrical fault is simulated in the zone overlap region and the associated underground ACBs are verified to operate correctly. This surveillance is required on a 12 month Frequency. The 12 month Frequency is based on engineering judgement and provides reasonable assurance that the zone overlap protection circuitry is operating properly.

This SR is modified by a Note indicating the SR is only applicable when the overhead disconnects to the underground KHU are closed. When the overhead disconnects to the underground KHU are open, the circuitry preventing the zone overlap protective lockout of both KHUs is not needed.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.14

This surveillance verifies OPERABILITY of the trip functions of the SL and N breakers. This SR verifies each trip circuit of each breaker independently opens each breaker. Neither of these breakers have any automatic close functions; therefore, only the trip circuits require verification. The 18 month Frequency is based on engineering judgement and provides reasonable assurance that the SL and N breakers will trip when required.

The SR is modified by a Note indicating that the SR is not required for an SL breaker when its standby bus is energized by a LCT via an isolated power path. This is necessary since the standby buses are required to be energized from a LCT by several Required Actions of Specification 3.8.1 and the breakers must remain closed to energized the standby buses from a LCT.

SR 3.8.1.15

This surveillance verifies proper operation of the 230 kV switchyard circuit breakers upon an actual or simulated actuation of the Switchyard Isolation circuitry. This test causes an actual switchyard isolation (by actuation of degraded grid voltage protection) and alignment of KHUs to the overhead and underground emergency power paths. An 18 month Frequency minimizes the impact to the Station and the operating Units which are connected to the 230 kV switchyard. The effect of this SR is not significant because the generator red bus tie breakers and feeders from the Oconee 230 kV switchyard red bus to the system grid remain closed. Either Switchyard Isolation Channel causes full system realignment, which involves a complete switchyard realignment. To avoid excessive switchyard circuit breaker cycling, realignment and KHU emergency start functions, this SR need be performed only once each SR interval.

This SR is modified by a Note. This Note states the redundant breaker trip coils shall be verified on a STAGGERED TEST BASIS. Verifying the trip coils on a STAGGERED TEST BASIS precludes unnecessary breaker operation

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.15 (continued)

and minimizes the impact to the Station and the operating Units which are connected to the 230 kV switchyard.

SR 3.8.1.16

This SR verifies by administrative means that one KHU provides an alternate manual AC power source capability by manual or automatic KHU start with manual synchronize, or breaker closure, to energize its non-required emergency power path. That is, when the KHU to the overhead emergency power path is inoperable, the SR verifies by administrative means that the overhead emergency power path is OPERABLE. When the overhead emergency power path is inoperable, the SR verifies by administrative means that the KHU associated with the overhead emergency power path is OPERABLE.

This SR is modified by a Note indicating that the SR is only applicable when complying with Required Action C.2.2.4.

REFERENCES

1. UFSAR, Section 3.1.39
2. UFSAR, Chapter 16
3. 10 CFR 50.36
4. UFSAR, Chapter 6
5. UFSAR, Chapter 15
6. Regulatory Guide 1.32
7. Regulatory Guide 1.129
8. IEEE-450-1980
9. UFSAR, Section 6.3.3.3

BASES

LCO (continued)

switchyard available or connected through the 100 kV line and transformer CT-5 to one main feeder bus.

In MODES 5 or 6 and during movement of irradiated fuel, a Lee Combustion Turbine (LCT) energizing one standby bus via an isolated power path to one main feeder bus can be utilized as an emergency power source. The LCT is required to provide power within limits of voltage and frequency using the 100 kV transmission line electrically separated from the system grid and offsite loads energizing one or more standby buses through transformer CT-5. The required number of energized standby buses is based upon the requirements of LCO 3.8.9, "Distribution System - Shutdown."

An OPERABLE KHU must be capable of starting, accelerating to rated speed and voltage, and connecting to the main feeder bus(es). The sequence must be capable of being accomplished within 23 seconds after a manual emergency start initiation signal. An emergency power source must be capable of accepting required loads and must continue to operate until offsite power can be restored to the main feeder buses.

This LCO is modified by three Notes. Note 1 indicates that a unit startup transformer may be shared with a unit in MODES 5 and 6. Note 2 indicates that the requirements of Specification 5.5.19, "Lee Combustion Turbine Testing Program," shall be met when a Lee Combustion Turbine (LCT) is used for the emergency power requirements. Note 3 indicates that the required emergency power source and the required offsite power source shall not be susceptible to a failure disabling both sources.

The required emergency power source and required offsite source cannot be susceptible to a failure disabling both sources. If the required offsite source is the 230 kV switchyard and the startup transformer energizing the required main feeder bus(es), the KHU and its required underground emergency power path are required to be OPERABLE since it is not subject to a failure, such as an inoperable startup transformer, which simultaneously disables the offsite source. If the Central switchyard is serving as the required offsite source through the CT-5 transformer with a power path through only one standby bus, the KHU and its required underground emergency power path cannot be used as the emergency power source if the power path is through the

(continued)

BASES

LCO (continued)

same standby bus since a single failure of a standby bus would disable both sources. Conversely, if an LCT is being used as an emergency power source, the required offsite source must be an offsite circuit available or connected through the startup transformer or a backcharged unit main step-up transformer and the unit auxiliary transformer.

APPLICABILITY

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

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

b. Systems needed to mitigate a fuel handling accident 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

A.1

An offsite source would be considered inoperable if it were not available to one required main feeder bus. Although two main feeder buses may be required by LCO 3.8.9, the one main feeder bus with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By the allowance of the option to declare features inoperable with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

(continued)

BASES

BACKGROUND (continued)

three units. The ASW pump suction supply is lake water from the embedded Unit 2 condenser circulating water (CCW) piping.

The SSF ASW System is used to provide adequate cooling to maintain single phase RCS natural circulation flow in MODE 3 with an average RCS temperature ≥ 525°F (unless the initiating event causes the unit to be driven to a lower temperature). In order to maintain single phase RCS natural circulation flow, at least 5 of the 9 Bank 2, Group B pressurizer heaters must be OPERABLE. These heaters are needed to compensate for ambient heat loss from the pressurizer. As long as the temperature in the pressurizer is maintained, RCS pressure will also be maintained. This will preclude hot leg voiding and ensure adequate natural circulation cooling.

The SSF Portable Pumping System, which includes a submersible pump and a flow path capable of taking suction from the intake canal and discharging into the Unit 2 CCW line, is designed to provide a backup supply of water to the SSF in the event of loss of CCW and subsequent loss of CCW siphon flow. The SSF Portable Pumping System is installed manually according to procedures.

The SSF RC Makeup System is designed to supply makeup to the RCS in the event that normal makeup systems are unavailable. An SSF RC Makeup Pump located in the Reactor Building of each unit supplies makeup to the RCS should the normal makeup system flow and seal cooling become unavailable. The system is designed to ensure that sufficient borated water is provided from the spent fuel pools to allow the SSF to maintain all three units in MODE 3 with average RCS temperature ≥ 525°F (unless the initiating event causes the unit to be driven to a lower temperature) for approximately 72 hours. An SSF RC Makeup Pump is capable of delivering borated water from the Spent Fuel Pool to the RC pump seal injection lines. A portion of this seal injection flow is used to makeup for reactor coolant pump seal leakage while the remainder flows into the RCS to makeup for other RCS leakage (non LOCA).

The SSF Power System includes 4160 VAC, 600 VAC, 208 VAC, 120 VAC and 125 VDC power. It consists of switchgear, a load center, motor control centers, panelboards, remote starters, batteries, battery chargers, inverters, a diesel

(continued)

BACKGROUND (continued)

generator (DG), relays, control devices, and interconnecting cable supplying the appropriate loads.

The AC power system consists of 4160 V switchgear OTS1; 600 V load center OXSF; 600 V motor control centers XSF, 1XSF, 2XSF, 3XSF; 208 V motor control centers 1XSF, 1XSF-1, 2XSF, 2XSF-1, 3XSF, 3XSF-1; 120 V panelboards KSF, KSFC.

The DC power system consists of two 125 VDC batteries and associated chargers, two 125 VDC distribution centers (DCSF, DCSF-1), and a DC power panelboard (DCSF). Only one battery and associated charger is required to be operable and connected to the 125 VDC distribution center to supply the 125 VDC loads. In this alignment, which is normal, the battery is floated on the distribution center and is available to assure power without interruption upon loss of its associated battery charger or AC power source. The other 125 VDC battery and its associated charger are in a standby mode and are not normally connected to the 125 VDC distribution center. However, they are available via manual connection to the 125 VDC distribution center to supply SSF loads, if required.

The SSF Power System is provided with standby power from a dedicated DG. The SSF DG and support systems consists of the diesel generator, fuel oil transfer system, air start system, diesel engine service water system, as well as associated controls and instrumentation. This SSF DG is rated for continuous operation at 3500 kW, 0.8 pf, and 4160 VAC. The SSF electrical design load does not exceed the continuous rating of the DG. The auxiliaries required to assure proper operation of the SSF DG are supplied entirely from the SSF Power System. The SSF DG is provided with manual start capability from the SSF only. It uses a compressed air starting system with four air storage tanks. An independent fuel system, complete with a separate underground storage tank, duplex filter arrangement, a fuel oil transfer pump, and a day tank, is supplied for the DG.

The SSF Power System OPERABILITY is supported by portions of the SSF HVAC System, consisting of the SSF Air Conditioning (AC) and Ventilation Systems. The SSF AC System, which includes the HVAC service water system and AC equipment (fan motors, compressors, condensors, and coils) must be OPERABLE to support SSF Power System OPERABILITY. The SSF AC System is designed to maintain the SSF Control Room, Computer Room,

(continued)

BASES

BACKGROUND (continued)

and Battery Rooms within their design temperature range. Elevated temperatures in the SSF Control Room and Computer Room could cause the SSF Power System to fail during an accident which requires operation of the SSF. Since the SSF HVAC service water pumps perform a redundant function, only one of two are required to be OPERABLE for the SSF HVAC service water system to be considered OPERABLE. The SSF Ventilation System, which supplies outside air to the Switchgear, Pump, HVAC and Diesel Generator Rooms, is composed of the following four subsystems: Constant Ventilation, Summer Ventilation, On-line Ventilation, and Diesel Generator Engine Ventilation. These ventilation systems work together to provide cooling to the various rooms of the SSF under both standby and on-line modes. The Diesel Generator Engine Ventilation fan is required for OPERABILITY of the SSF Power System. The six fans associated with the other three ventilation systems may or may not be required for SSF OPERABILITY dependent upon outside air temperature. If one of these ventilation fans fail, an engineering evaluation must be performed to determine if any of the SSF Systems or instrumentation are inoperable.

SSF Instrumentation is provided to monitor RCS pressure, RCS Loop A and B temperature (hot leg and cold leg), pressurizer water level, and SG A and B water level. Indication is displayed on the SSF control panel.

APPLICABLE SAFETY ANALYSES

The SSF serves as a backup for existing safety systems to provide an alternate and independent means to achieve and maintain one, two, or three Oconee units in MODE 3 with average RCS temperature ≥ 525°F (unless the initiating event causes the unit to be driven to a lower temperature) for up to 72 hours following 10 CFR 50 Appendix R fire, a turbine building flood, sabotage, SBO, or tornado missile events (Refs. 1, 6, 7, and 8).

The OPERABILITY of the SSF is consistent with the assumptions of the Oconee Probabilistic Risk Assessment (Ref. 2). Therefore, the SSF satisfies Criterion 4 of 10 CFR 50.36 (Ref. 3).

(continued)

BASES (continued)

LCO

The SSF Instrumentation in Table B 3.10.1-1 and the following SSF Systems shall be OPERABLE:

a. SSF Auxiliary Service Water System;

b. SSF Portable Pumping System;

c. SSF Reactor Coolant Makeup System; and

d. SSF Power System.

An OPERABLE SSF ASW System includes five pressurizer heaters capable of being powered from the SSF, and an SSF ASW pump, piping, instruments, and controls to ensure a flow path capable of taking suction from the Unit 2 condenser circulating water (CCW) line and discharging into the secondary side of each SG. An OPERABLE SSF Portable Pumping System includes an SSF submersible pump and a flow path capable of taking suction from the intake canal and discharging into the Unit 2 CCW line. An OPERABLE Reactor Coolant Makeup System includes an SSF RC makeup pump, piping, instruments, and controls to ensure a flow path capable of taking suction from the spent fuel pool and discharging into the RCS. An OPERABLE SSF Power System includes the SSF DG, diesel support systems, 4160 VAC, 600 VAC, 208 VAC, 120 VAC, and 125 VDC systems. Only one 125 VDC SSF battery and its associated charger are required to be OPERABLE to support OPERABILITY of the 125 VDC system.

APPLICABILITY

The SSF System is required in MODES 1, 2, and 3 to provide an alternate means to achieve and maintain the unit in MODE 3 with average RCS temperature ≥ 525°F (unless the initiating event causes the unit to be driven to a lower temperature) following 10 CFR 50 Appendix R fire, turbine building flood, sabotage, SBO and tornado missile events. The safety function of the SSF is to achieve and maintain the unit in MODE 3 with average RCS temperature ≥ 525°F (unless the initiating event causes the unit to be driven to a lower temperature); therefore, this LCO is not applicable in MODES 4, 5, or 6.

(continued)

BASES (continued)

ACTIONS

The exception for LCO 3.0.4, provided in the Note of the Actions, permits entry into MODES 1, 2, and 3 with the SSF not OPERABLE. This is acceptable because the SSF is not required to support normal operation of the facility or to mitigate a design basis accident.

A.1, B.1, C.1, D.1, and E.1

With one or more of the SSF Systems inoperable or the required SSF instrumentation of Table B 3.10.1-1 inoperable, the SSF is in a degraded condition and the system(s) or instrumentation must be restored to OPERABLE status within 7 days. The 7 day Completion Time is based on the low probability of an event occurring which would require the SSF to be utilized.

F.1

If the Required Action and associated Completion Time of Condition A, B, C, D, or E are not met when SSF Systems or Instrumentation are inoperable due to maintenance, the unit may continue to operate provided that the SSF is restored to OPERABLE status within 45 days from discovery of initial inoperability.

This Completion Time is modified by a Note that indicates that the SSF shall not be in Condition F for more than a total of 45 days in a calendar year. This includes the 7 day Completion Time that leads to entry into Condition F. For example, if the SSF ASW System is inoperable for 10 days, the 45 day special inoperability period is reduced to 35 days. If the SSF ASW System is inoperable for 6 days, Condition A applies and there is no reduction in the 45 day allowance. The limit of 45 days per calendar year minimizes the number and duration of extended outages associated with exceeding the 7 day Completion Time of a Condition.

G.1 and G.2

If the Required Action and associated Completion Time of Condition F are not met or if the Required Action and associated Completion Time of Condition A, B, C, D, or E are not met for reasons other than Condition F, the unit must be

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and MODE 4 within 84 hours. The allowed Completion Times are appropriate, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems, considering a three unit shutdown may be required.

SURVEILLANCE REQUIREMENTS

SR 3.10.1.1

Performance of the CHANNEL CHECK once every 7 days for each required instrumentation channel ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel with 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; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. This SR is modified by a Note to indicate that it is not applicable to the SSF RCS temperature instrument channels, which are common to the RPS RCS temperature instrument channels and are normally aligned through a transfer isolation device to each Unit control room. The instrument string to the SSF control room is checked and calibrated every 18 months.

Agreement criteria are determined 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. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.10.1.1 (continued)

verified to be reading at the bottom of the range and not failed downscale.

The Frequency is based on unit operating experience that demonstrates channel failure is rare.

SR 3.10.1.2

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 4).

SR 3.10.1.3 and 3.10.1.4

SR 3.10.1.3 provides verification that the level of fuel oil in the day tank is at or above the level at which fuel oil is automatically added. The level is expressed as an equivalent volume in gallons. The day tank is sized based on the amount of fuel oil required to successfully start the DG and to allow for orderly shutdown of the DG upon loss of fuel oil from the main storage tank.

SR 3.10.1.4 provides verification that there is an adequate inventory of fuel oil in the storage tanks to support SSF DG operation for 72 hours at full load. The 72 hour period is sufficient time to place the unit in a safe shutdown condition.

The 31 day Frequency for these SRs is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.1.5

The SR requires the DG to start (normal or emergency) from standby conditions and achieve required voltage and frequency. Standby conditions for a DG means that the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations. This SR is modified by a Note to indicate that all DG starts for this Surveillance may be preceded by an engine prelube period and followed by a warmup period prior to loading. This minimizes wear on oving parts that do not get lubricated when the engine is running.

The 31 day Frequency is consistent with Regulatory Guide 1.9 (Ref. 5). This Frequency provides adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

SR 3.10.1.6

This Surveillance ensures that sufficient air start capacity for the SSF DG is available, without the aid of the refill compressor. The SSF DG air start system is equipped with four air storage tanks. Each set of two tanks will provide sufficient air to start the SSF DG a minimum of three successive times without recharging. The pressure specified in this SR is intended to reflect the lowest value at which the three starts can be accomplished.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources.

SR 3.10.1.7

This Surveillance demonstrates that the fuel oil transfer pump automatically starts and transfers fuel oil from the underground fuel oil storage tank to the day tank. This is required to support continuous operation of SSF DG. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.10.1.7 (continued)

The 92 day Frequency is considered acceptable based on operating experience.

SR 3.10.1.8

A sample of fuel oil is required to be obtained from the SSF day tank and underground fuel oil storage tank in accordance ith the Diesel Fuel Oil Testing Program in order to ensure that fuel oil viscosity, water, and sediment are within the limits of the Diesel Fuel Oil Testing Program.

The 92 day Frequency is considered acceptable based on operating experience related to diesel fuel oil quality.

SR 3.10.1.9

This Surveillance verifies that the SSF DG is capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize electrical loads, while minimizing the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while the 1.0 is an operational limitation to ensure circulating currents are minimized. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

The normal 92 day Frequency for this Surveillance is consistent with Regulatory Guide 1.9 (Ref. 3).

This SR is modified by three Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients because

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.10.1.9 (continued)

of changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit will not invalidate the test. Note 3 indicates that all DG starts for this Surveillance may be preceded by an engine prelube period and followed by a warmup period prior to loading. This minimizes wear on moving parts that do not get lubricated.

SR 3.10.1.10

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The presence of physical damage or deterioration does not necessarily represent a failure of this SR, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function).

The 12 month Frequency for this SR is consistent with IEEE-450 (Ref. 4), which recommends detailed visual inspection of cell condition and rack integrity on a yearly basis.

SR 3.10.1.11

Visual inspection of battery cell to cell and terminal connections provides an indication of physical damage that could potentially degrade battery performance. The anti-corrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection.

The limits established for this SR must be no more than 20% above the resistance as measured during installation or not above the ceiling value established by the manufacturer.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.10.1.11 (continued)

The Surveillance Frequency for these inspections is 12 months. This Frequency is considered acceptable based on operating experience related to detecting corrosion trends.

SR 3.10.1.12

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length correspond to the design duty cycle requirements. The design basis discharge time for the SSF battery is one hour.

The Surveillance Frequency for this test is 12 months. This Frequency is considered acceptable based on operating experience.

SR 3.10.1.13

CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis. This Frequency is justified by the assumption of an 18 month calibration interval to determine the magnitude of equipment drift in the setpoint analysis.

SR 3.10.1.14

Inservice Testing of the SSF valves demonstrates that the valves are mechanically OPERABLE and will operate when required. These valves are required to operate to ensure the required flow path.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.10.1.14 (continued)

The specified Frequency is in accordance with the IST Program requirements. Operating experience has shown that these components usually pass the SR when performed at the IST Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.10.1.15

This SR requires the SSF pumps to be tested in accordance with the IST Program. The IST verifies the required flow rate at a discharge pressure to verify OPERABILITY. The SR is modified by a note indicating that it is not applicable to the SSF submersible pump.

The specified Frequency is in accordance with the IST Program requirements. Operating experience has shown that these components usually pass the SR when performed at the IST Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.10.1.16

This SR requires the SSF submersible pump to be tested on a 2 year Frequency and verifies the required flow rate at a discharge pressure to verify OPERABILITY.

The specified Frequency is based on the pump being not QA grade and on operating experience that has shown it usually passes the SR when performed at the 2 year Frequency.

REFERENCES

1. UFSAR, Section 9.6.
2. Oconee Probabilistic Risk Assessment.
3. 10 CFR 50.36.
4. IEEE-450-1987.
5. Regulatory Guide 1.9, Rev. 0, December 1974.

(continued)

BASES

REFERENCES (continued)

6. NRC Letter from L. A. Wiens to H. B. Tucker, "Safety Evaluation Report on Effect of Tornado Missiles on Oconee Emergency Feedwater System," dated July 28, 1989.

7. NRC Letter from L. A. Wiens to J. W. Hampton, "Safety Evaluation for Station Blackout (10 CFR 50.63) - Oconee Nuclear Station, Units 1, 2, and 3," dated March 10, 1992.

8. NRC Letter from L. A. Wiens to J. W. Hampton, "Supplemental Safety Evaluation for Station Blackout (10 CFR 50.63) - Oconee Nuclear Station, Units 1, 2, and 3," dated December 10, 1992.