Atmospheric Dump Valve (ADV) Flow Paths

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	-------------------NOTE------------------- With three RCPs operating, the limits are applied to the loop with the highest pressure. -- Verify RCS loop pressure is within limits specified in the COLR.	12 hours
SR 3.4.1.2	-------------------NOTE------------------- With three RCPs operating, the limits are applied to the loop with the lowest loop average temperature. for the condition where there is a 0°F ΔTc setpoint. -- Verify RCS loop average temperature is within limits specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow is within limits specified in the COLR.	12 hours
SR 3.4.1.4	-------------------NOTE------------------- Not required to be performed until 7 days after stable thermal conditions are established in the higher power range of MODE 1. -- Verify by measurement RCS total flow rate is within limit specified in the COLR.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

2425

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE with lift settings ≥ 2475 psig and ≤ 2525 psig.

2575

APPLICABILITY: MODES 1 and 2,
MODE 3 with all RCS cold leg temperatures > 325°F.

-----NOTE-----

The lift settings are not required to be within the LCO limits for entry into the applicable portions of MODE 3 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 36 hours following entry into the applicable portions of MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 3 with any RCS cold leg temperature ≤ 325°F.	12 hours 18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program

Following testing, lift settings shall be within ±1%.

3.7 PLANT SYSTEMS

3.7.4 ~~Not used~~ Atmospheric Dump Valve (ADV) Flow Paths

Insert Proposed Technical Specification 3.7.4

INSERT PROPOSED SPECIFICATION 3.9.7

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. Nuclear Overpower Flux/Flow/Imbalance and RCS Variable Low Pressure allowable value limits for Specification 3.3.1;

7. RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits for Specification 3.4.1

8. Core Flood Tanks Boron concentration limits for Specification 3.5.1;

9. Borated Water Storage Tank Boron concentration limits for Specification 3.5.4;

10. Spent Fuel Pool Boron concentration limits for Specification 3.7.12;

11. RCS and Transfer Canal boron concentration limits for Specification 3.9.1; and

12. AXIAL POWER IMBALANCE protective limits and RCS Variable Low Pressure protective limits for Specification 2.1.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

(1) DPC-NE-1002A, Reload Design Methodology II, ~~October 1985~~; Rev. 1, (SER dated October 1, 1985)

(2) NFS-1001A, Reload Design Methodology, ~~April 1984~~; Rev. 4, (SER dated July 29, 1981)

(3) DPC-NE-2003P-A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01, ~~July 1989~~; (SER dated July 19, 1989)

(4) DPC-NE-1004A, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, ~~November 1992~~; (SER dated November 23, 1992)

~~(5) BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November, 1989;~~

~~(6) BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, as approved by SER dated February, 1994;~~

(7) DPC-NE-3000P-A, Thermal Hydraulic Transient Analysis Methodology, ~~August, 1994~~; Rev. 2, (SER dated October 14, 1998

(5) DPC-NE-2008P-A, Fuel Mechanical Reload Analysis Methodology Using TACO3, (SER dated April 3, 1995).

(6) BAW-10192-PA, BWNT LOCA - BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants, (SER dated February 18, 1997)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

(8) DPC-NE-2005P-A, Thermal Hydraulic Statistical Core Design Methodology, ~~February, 1995~~ and Rev. 1, (SER dated November 7, 1996)

A

(9) DPC-NE-3005-P, UFSAR Chapter 15 Transient Analysis Methodology, ~~DPC, JUL97~~. Rev. 1, (SER dated May 25, 1999)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) and Main Feeder Bus Monitor Panel (MFPMP) Report

When a report is required by Condition B or G of LCO 3.3.8, "Post Accident Monitoring (PAM) Instrumentation" or Condition D of LCO 3.3.23, "Main Feeder Bus Monitor Panel," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring (PAM only), the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

The steam generator tube inspection report shall comply with the following:

a. The number of tubes plugged or repaired in each steam generator shall be reported to the NRC within 30 days following the completion of the plugging or repair procedure.

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BASES

ACTIONS (continued)

A.2.4

0.2% Δk/k at RTP, 0.4% at 80% RTP, or 0.8% Δk/k at zero power

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of ~~0.65% Δk/k at RTP or 1.0% Δk/k at zero power~~ (Ref. 3). This evaluation may require a computer calculation of the maximum ejected rod worth based on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the duration of time that operation is expected to continue with a misaligned rod. Should fuel cycle conditions at some later time become more bounding than those at the time of the rod misalignment, additional evaluation will be required to verify the continued acceptability of operation. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation.

B.1

If the Required Actions and associated Completion Times for Condition A 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. The allowed Completion Time of 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

C.1.1

More than one trippable CONTROL ROD becoming inoperable or misaligned, or both inoperable but trippable and misaligned from their group average position, is not expected and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour allows the operator adequate time to determine the SDM.

C.1.2

If the SDM is less than the limit, then the restoration of the required SDM requires increasing the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. RCS HIGH PRESSURE (continued)

The RCS High Pressure trip has been credited in the transient analysis calculations for slow positive reactivity insertion transients (rod withdrawal transients and moderator dilution). The rod withdrawal transient covers a large spectrum of reactivity insertion rates and rod worths that exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower – High Setpoint trip provides the primary protection. At low reactivity insertion rates, the RCS High Pressure trip provides the primary protection.

The setpoint Allowable Value is selected to ensure that the RCS High Pressure SL is not challenged during steady state operation or slow power increasing transients. The setpoint Allowable Value does not reflect errors induced by harsh environmental conditions because the equipment is not required to mitigate accidents that create harsh conditions in the RB.

4. RCS Low Pressure

The RCS Low Pressure trip, in conjunction with the RCS High Outlet Temperature and Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system pressure approaches the conditions necessary for DNB. The RCS Low Pressure trip provides DNB low pressure limit for the RCS Variable Low Pressure trip.

The RCS Low Pressure setpoint Allowable Value is selected to ensure that a reactor trip occurs before RCS pressure is reduced below the lowest point at which the RCS Variable Low Pressure trip is analyzed. The RCS Low Pressure trip provides protection for primary system depressurization events and has been credited in the accident analysis calculations for small break loss of coolant accidents (LOCAs) and main steam line break (MSLB) accidents. Harsh RB conditions created by small break LOCAs cannot affect performance of the RCS pressure sensors and transmitters within the time frame for a reactor trip. Therefore, degraded environmental conditions are not considered in the Allowable Value determination.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5. RCS Variable Low Pressure

The RCS Variable Low Pressure trip, in conjunction with the RCS High Outlet Temperature and RCS Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system parameters of pressure and temperature approach the conditions necessary for DNB. The RCS Variable Low Pressure trip provides a floating low pressure trip based on the RCS High Outlet Temperature within the range specified by the RCS High Outlet Temperature and RCS Low Pressure trips.

The RCS Variable Low Pressure setpoint Allowable Value is selected to ensure that a trip occurs when temperature and pressure approach the conditions necessary for DNB while operating in a temperature pressure region constrained by the low pressure and high temperature trips. The RCS Variable Low Pressure trip is assumed for transient protection in the ~~unit safety analysis but does not affect the limiting cases; therefore, determination of the setpoint Allowable Value does not account for errors induced by a harsh RB environment.~~

main steam line break analysis. The setpoint allowable valve does not include errors induced by the harsh environment, because the trip actuates prior to the harsh environment.

6. Reactor Building High Pressure

The Reactor Building High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB. By detecting changes in the RB pressure, the RPS can provide a reactor trip before the other system parameters have varied significantly. Thus, this trip acts to minimize accident consequences. It also provides a backup for RPS trip instruments exposed to an RB HELB environment.

The Allowable Value for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. The electronic components of the RB High Pressure trip are located in an area that is not exposed to high temperature steam environments during HELB transients inside containment. The components are exposed to high radiation conditions. Therefore, the determination of the setpoint Allowable Value accounts for errors induced by the high radiation.

B 3.3 INSTRUMENTATION

B 3.3.15 Turbine Stop Valve (TSV) Closure

BASES

BACKGROUND

The Turbine Stop Valves (TSV) Closure function partially isolates the main steam lines from the SGs by closing the TSVs on both main steam lines following a high energy line break (HELB). turbine or reactor trip signal

Two TSVs are provided for each main steam line and are located outside of containment. The TSVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent the MSSVs and EFW pump's steam supply from being isolated from the steam generators by TSV closure. Closing the TSVs partially isolates each steam generator from the other, and isolates the turbine from the steam generators.

TSV Closure is initiated by a reactor trip. To keep from rapidly cooling down the primary plant by drawing off too much steam, the turbine is tripped when the reactor trips. Two independent and redundant "Reactor Trip Confirmed" signals in the form of contact closures from the control rod drive system will energize two independent turbine trip mechanisms. The Channel A trip circuit will close all four TSVs within a maximum of 1 second. The Channel B trip circuit will close the TSVs within a maximum of 15 seconds.

APPLICABLE SAFETY ANALYSES

The design basis of the TSV Closure function is established by the analysis for the main steam line break (MSLB) as discussed in the UFSAR, Section 15.13 (Ref. 1). TSV closure is necessary to stop steam flow to the turbine (to prevent overcooling) following all reactor trips.

The MSLB with ICS low level control and without operator action prior to ten minutes is the limiting case for a post-trip return to power.

The accident analysis compares several different MSLB events. The MSLB outside containment upstream of the TSV is limiting for offsite dose, although a ~~break in this section of main steam header~~ has a very low probability. ~~The main MSLB without ICS and without operator action is the limiting case for a post trip return to power.~~ The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed for include loss of coolant flow events and dropped control rod events and control rod withdrawal events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.2.1, "Regulating Rod Position Limits," LCO 3.2.2, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.3, "QUADRANT POWER TILT (QPT)."

The normal operating band for RCS pressure is between 2125 psig and 2155 psig as measured at the hot leg pressure tap. The safety analyses assume a core exit pressure that is based on the measured pressure and concurrent pressure losses between the two locations. The pressure losses are a function of the loop flow rate, thus different values are allowed for 4 or 3 RCP operation.

Analyses have been performed to establish the pressure, temperature, and flow rate requirements for three pump and four pump operation. These limits are specified in the COLR. The flow limits for three pump operation are substantially lower than for four pump operation. To meet the DNBR criterion, a corresponding maximum power limit is required (see Bases for LCO 3.4.4, "RCS Loops–MODES 1 and 2").

Another set of limits on DNBR related parameters is provided in Safety Limit (SL) 2.1.1, "Reactor Core SLs." Those limits are less restrictive than the limits of LCO 3.4.1, but violation of an SL merits a stricter, more severe Required Action.

The RCS DNB limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

This LCO specifies limits on the monitored process variables: RCS loop (hot leg) pressure, RCS loop average temperature, and RCS total flow rate to ensure that the core operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting DNBR criteria in the event of a DNB limited transient.

The pressure limits are applied to the loop with the highest pressure. The temperature limits are applied to the loop with the lowest loop average temperature.

for the condition in which there is a 0°F ΔT_c setpoint

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 (continued)

restored to a normal operation, steady state condition following load changes and other expected transient operations. The RCS pressure value specified in the COLR is dependent on the number of pumps in operation and has been adjusted to account for the pressure loss difference between the core exit and the measurement location. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation is within safety analysis assumptions. A Note has been added to indicate the pressure limits for three pumps operating is applied to the loop with the highest pressure.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for loop average temperature is sufficient to ensure that the RCS coolant temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions. A Note has been added to indicate the temperature limits for three pumps operating are applied to the loop with the lowest loop average temperature.

for the condition in which there is a 0°F ΔT_c setpoint.

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a calorimetric heat balance once every 18 months allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow is greater than or equal to the minimum required RCS flow rate specified in the COLR.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the two spring loaded pressurizer safety valves is to provide RCS overpressure protection. Operating in conjunction with the Reactor Protection System (RPS), two valves are used to ensure that the Safety Limit (SL) of 2750 psig is not exceeded for analyzed transients during operation in MODES 1 and 2. Two safety valves are used for portions of MODE 3. For the remainder of MODE 3, MODE 4, MODE 5, and MODE 6 with the reactor head on, overpressure protection is provided by operating procedures and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III (Ref. 1). The setpoint of the pressurizer code safety valves is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967. ~~The required lift pressure is 2500 psig ± 1%.~~ The safety valves discharge steam from the pressurizer to a quench tank located in the containment. The discharge flow is indicated by an increase in temperature downstream of the safety valves and by an increase in the quench tank temperature and level.

Insert A

~~The upper and lower pressure limits are based on the ± 1% tolerance requirement for lifting pressures above 1000 psig.~~ The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the ASME pressure limit could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

All accident analyses in the UFSAR that require safety valve actuation assume operation of both pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis is also

Insert A

The required lift pressure is 2500 psig +/- 3%. The upper and lower pressure limits are based on the requirements of ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967, which limit the rise in pressure within the vessels which they protect to 10% above the design pressure.

BASES

APPLICABLE SAFETY ANALYSES (continued)

3

based on operation of both safety valves and assumes that the valves open at the high range of the setting (2500 psig system design pressure plus 1%). These valves must accommodate pressurizer insurges that could occur during a startup, rod withdrawal, ejected rod, or loss of main feedwater. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at < 15% power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this Specification is required to ensure that the accident analysis and design basis calculations remain valid.

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

INSERT B

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psig) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions and to comply with ASME Code requirements. ~~The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure.~~ Inoperability of one or both valves could result in exceeding the SL if a transient were to occur.

The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and portions of MODE 3 above the LTOP cut in temperature, OPERABILITY of two valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. Portions of MODE 3 are conservatively included, although the listed accidents may not require both safety valves for protection.

The LCO is not applicable in MODE 3 when any RCS cold leg temperature is ≤ 325°F, MODE 4 and MODE 5 because LTOP protection is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head detensioned.

Insert B

The valves will be tested per ASME Section XI requirements and returned to service with as-left setpoints of 2500 psig +/- 1%. The upper and lower pressure tolerance limits are based on the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967, which limits the rise in pressure within the vessel which they protect, to 10% above the design pressure.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 2), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valves setpoint is ±3% for OPERABILITY; however, the valves are reset to ±1% during the Surveillance to allow for drift. These values include instrument uncertainties.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. ASME, Boiler and Pressure Vessel Code, Section XI.
3. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.2 Turbine Stop Valves (TSVs)

BASES

BACKGROUND

The TSVs partially isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). TSV closure partially terminates flow from the unaffected (intact) steam generator.

Two TSVs are provided for each main steam line and are located outside of containment. The TSVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent the MSSVs and EFW pump's steam supply from being isolated from the steam generators by TSV closure. Closing the TSVs partially isolates each steam generator from the other, and isolates the turbine from the steam generators.

TSV Closure is initiated by a reactor trip. To keep from rapidly cooling down the primary plant by drawing off too much steam, the turbine is tripped when the reactor trips. Two independent and redundant "Reactor Trip Confirmed" signals in the form of contact closures from the control rod drive system will energize two independent turbine trip mechanisms. The Channel A trip circuit will close all four TSVs within a maximum of 1 second. The Channel B trip circuit will close the TSVs within a maximum of 15 seconds.

A discussion of the TSV's function is found in the UFSAR, Section 10.3 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the TSVs is established by the analysis for the main steam line break (MSLB) as discussed in the UFSAR, Section 15.13 (Ref. 2). TSV closure is necessary to stop steam flow to the turbine (to prevent overcooling) following all reactor trips. Another failure considered is the loss of one switchgear.

The MSLB with ICS low level control and no operator action prior to ten minutes is the limiting case for a post-trip return to power.

The accident analysis compares several different MSLB events. The main SLB outside containment upstream of the TSV is limiting for offsite dose. ~~The main SLB without ICS and operator Action is the limiting case for a post trip return to power.~~ With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed.

B 3.7 PLANT SYSTEMS

B 3.7.4 ~~Not used~~ Atmospheric Dump Valve (ADV) Flow Paths

Insert Proposed Bases for Technical Specification 3.7.4

Attachment 3

Retyped Technical Specifications and Bases

Instructions for Updating

Technical Specifications and Bases

Remove Page	**Insert Page**
Technical Specifications, Table of Contents, pages iii and iv	Technical Specifications, Table of Contents, pages iii and iv
3.4.1-2	3.4.1-2
3.4.10-1 - 3.4.10-2	3.4.10-1 - 3.4.10-2
3.7.4-1	3.7.4-1
---------------	3.9.7-1 - 3.9.7-2
5.0-29 - 5.0-31	5.0-29 - 5.0-31
Bases, Table of Contents, pages iii and iv	Bases, Table of Contents, pages iii and iv
B 3.1.4-6	B 3.1.4-6
B 3.3.1-13 - B 3.3.3-14	B 3.3.1-13 - B 3.3.1-14
B 3.3.15-1	B 3.3.15-1
B 3.4.1-2	B 3.4.1-2
B 3.4.1-4	B 3.4.1-4
B 3.4.10-1 - B 3.4-10-4	B 3.4.10-1 - B 3.4.10-4
B 3.7.2-1	B 3.7.2-1
B 3.7.4-1	B 3.7.4-1 - B 3.7.4-3
--------------	B 3.9.7-1 - B 3.9.7-3

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	-------------------------NOTE------------------------ With three RCPs operating, the limits are applied to the loop with the highest pressure. --- Verify RCS loop pressure is within limits specified in the COLR.	 12 hours
SR 3.4.1.2	-------------------------NOTE------------------------ With three RCPs operating, the limits are applied to the loop with the lowest loop average temperature for the condition where there is a 0°F ΔTc setpoint. --- Verify RCS loop average temperature is within limits specified in the COLR.	 12 hours
SR 3.4.1.3	Verify RCS total flow is within limits specified in the COLR.	12 hours
SR 3.4.1.4	-------------------------NOTE------------------------ Not required to be performed until 7 days after stable thermal conditions are established in the higher power range of MODE 1. --- Verify by measurement RCS total flow rate is within limit specified in the COLR.	 18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE with lift settings ≥ 2425 psig and ≤ 2575 psig.

APPLICABILITY: MODES 1 and 2,
MODE 3 with all RCS cold leg temperatures > 325°F.

------------------------------NOTE------------------------------
The lift settings are not required to be within the LCO limits for entry into the applicable portions of MODE 3 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 36 hours following entry into the applicable portions of MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1	Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
<u>OR</u> Two pressurizer safety valves inoperable.	B.2	Be in MODE 3 with any RCS cold leg temperature ≤ 325°F.	18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Dump Valve (ADV) Flow Paths

LCO 3.7.4 The ADV flow path for each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3, and MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both ADV flow path(s) inoperable.	A.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	A.2 Be in MODE 4 without reliance upon steam generator for heat removal.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.4.1 Cycle the valves that comprise the ADV flow paths.	18 months

3.9 REFUELING OPERATIONS

3.9.7 Unborated Water Source Isolation Valves

LCO 3.9.7 Each valve used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 6.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each unborated water source isolation valve.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ------------NOTE----------- Required Action A.3 must be completed whenever Condition A is entered. ------------------------------ One or more valves not secured in closed position.	A.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	A.2	Initiate actions to secure valve in closed position.	Immediately
	<u>AND</u>		
	A.3	Perform SR 3.9.1.1.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify each valve that isolates unborated water sources is secured in the closed position.	31 days

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. Nuclear Overpower Flux/Flow/Imbalance and RCS Variable Low Pressure allowable value limits for Specification 3.3.1;

7. RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits for Specification 3.4.1

8. Core Flood Tanks Boron concentration limits for Specification 3.5.1;

9. Borated Water Storage Tank Boron concentration limits for Specification 3.5.4;

10. Spent Fuel Pool Boron concentration limits for Specification 3.7.12;

11. RCS and Transfer Canal boron concentration limits for Specification 3.9.1; and

12. AXIAL POWER IMBALANCE protective limits and RCS Variable Low Pressure protective limits for Specification 2.1.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

(1) DPC-NE-1002A, Reload Design Methodology II, Rev. 1, (SER dated October 1, 1985);

(2) NFS-1001A, Reload Design Methodology, Rev. 4, (SER dated July 29, 1981);

(3) DPC-NE-2003P-A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01, (SER dated July 19, 1989);

(4) DPC-NE-1004P-A, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, (SER dated November 23, 1992);

(5) DPC-NE-2008P-A, Fuel Mechanical Reload Analysis Methodology Using TACO3, (SER dated April 3, 1995);

(6) BAW-10192-PA, BWNT LOCA - BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants, (SER dated February 18, 1997);

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

(7) DPC-NE-3000P-A, Thermal Hydraulic Transient Analysis Methodology, Rev. 2, (SER dated October 14, 1998);

(8) DPC-NE-2005P-A, Thermal Hydraulic Statistical Core Design Methodology, Rev. 1, (SER dated November 7, 1996); and

(9) DPC-NE-3005-PA, UFSAR Chapter 15 Transient Analysis Methodology, Rev. 1, (SER dated May 25, 1999).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) and Main Feeder Bus Monitor Panel (MFPMP) Report

When a report is required by Condition B or G of LCO 3.3.8, "Post Accident Monitoring (PAM) Instrumentation" or Condition D of LCO 3.3.23, "Main Feeder Bus Monitor Panel," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring (PAM only), the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6 Reporting Requirements (continued)

5.6.8 Steam Generator Tube Inspection Report

The steam generator tube inspection report shall comply with the following:

a. The number of tubes plugged or repaired in each steam generator shall be reported to the NRC within 30 days following the completion of the plugging or repair procedure.

b. The results of the steam generator tube inservice inspection shall be reported to the NRC within 3 months following completion of the inspection. This report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of a degraded tube.

   3. Identification of tubes plugged or repaired.

   4. Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes.

c. Results of steam generator tube inspections which fall into Category C-3 and require notification to the NRC shall be reported prior to resumption of plant operation. The written report shall provide the results of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. The designation of affected and unaffected areas will be reported to the NRC when they are determined.

BASES

ACTIONS (continued)

A.2.4

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.2% $\Delta k/k$ at RPT, 0.4% $\Delta k/k$ at 80% RPT, or 0.8% $\Delta k/k$ at zero power. This evaluation may require a computer calculation of the maximum ejected rod worth based on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the duration of time that operation is expected to continue with a misaligned rod. Should fuel cycle conditions at some later time become more bounding than those at the time of the rod misalignment, additional evaluation will be required to verify the continued acceptability of operation. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation.

B.1

If the Required Actions and associated Completion Times for Condition A 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. The allowed Completion Time of 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

C.1.1

More than one trippable CONTROL ROD becoming inoperable or misaligned, or both inoperable but trippable and misaligned from their group average position, is not expected and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour allows the operator adequate time to determine the SDM.

C.1.2

If the SDM is less than the limit, then the restoration of the required SDM requires increasing the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. RCS HIGH PRESSURE (continued)

The RCS High Pressure trip has been credited in the transient analysis calculations for slow positive reactivity insertion transients (rod withdrawal transients and moderator dilution). The rod withdrawal transient covers a large spectrum of reactivity insertion rates and rod worths that exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower – High Setpoint trip provides the primary protection. At low reactivity insertion rates, the RCS High Pressure trip provides the primary protection.

The setpoint Allowable Value is selected to ensure that the RCS High Pressure SL is not challenged during steady state operation or slow power increasing transients. The setpoint Allowable Value does not reflect errors induced by harsh environmental conditions because the equipment is not required to mitigate accidents that create harsh conditions in the RB.

4. RCS Low Pressure

The RCS Low Pressure trip, in conjunction with the RCS High Outlet Temperature and Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system pressure approaches the conditions necessary for DNB. The RCS Low Pressure trip provides DNB low pressure limit for the RCS Variable Low Pressure trip.

The RCS Low Pressure setpoint Allowable Value is selected to ensure that a reactor trip occurs before RCS pressure is reduced below the lowest point at which the RCS Variable Low Pressure trip is analyzed. The RCS Low Pressure trip provides protection for primary system depressurization events and has been credited in the accident analysis calculations for small break loss of coolant accidents (LOCAs). Harsh RB conditions created by small break LOCAs cannot affect performance of the RCS pressure sensors and transmitters within the time frame for a reactor trip. Therefore, degraded environmental conditions are not considered in the Allowable Value determination.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5. RCS Variable Low Pressure

The RCS Variable Low Pressure trip, in conjunction with the RCS High Outlet Temperature and RCS Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system parameters of pressure and temperature approach the conditions necessary for DNB. The RCS Variable Low Pressure trip provides a floating low pressure trip based on the RCS High Outlet Temperature within the range specified by the RCS High Outlet Temperature and RCS Low Pressure trips.

The RCS Variable Low Pressure setpoint Allowable Value is selected to ensure that a trip occurs when temperature and pressure approach the conditions necessary for DNB while operating in a temperature pressure region constrained by the low pressure and high temperature trips. The RCS Variable Low Pressure trip is assumed for transient protection in the main steam line break analysis. The setpoint allowable value does not include errors induced by the harsh environment, because the trip actuates prior to the harsh environment.

6. Reactor Building High Pressure

The Reactor Building High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB. By detecting changes in the RB pressure, the RPS can provide a reactor trip before the other system parameters have varied significantly. Thus, this trip acts to minimize accident consequences. It also provides a backup for RPS trip instruments exposed to an RB HELB environment.

The Allowable Value for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. The electronic components of the RB High Pressure trip are located in an area that is not exposed to high temperature steam environments during HELB transients inside containment. The components are exposed to high radiation conditions. Therefore, the determination of the setpoint Allowable Value accounts for errors induced by the high radiation.

B 3.3 INSTRUMENTATION

B 3.3.15 Turbine Stop Valve (TSV) Closure

BASES

BACKGROUND

The Turbine Stop Valves (TSV) Closure function partially isolates the main steam lines from the SGs by closing the TSVs on both main steam lines following a turbine or reactor trip signal.

Two TSVs are provided for each main steam line and are located outside of containment. The TSVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent the MSSVs and EFW pump's steam supply from being isolated from the steam generators by TSV closure. Closing the TSVs partially isolates each steam generator from the other, and isolates the turbine from the steam generators.

TSV Closure is initiated by a reactor trip. To keep from rapidly cooling down the primary plant by drawing off too much steam, the turbine is tripped when the reactor trips. Two independent and redundant "Reactor Trip Confirmed" signals in the form of contact closures from the control rod drive system will energize two independent turbine trip mechanisms. The Channel A trip circuit will close all four TSVs within a maximum of 1 second. The Channel B trip circuit will close the TSVs within a maximum of 15 seconds.

APPLICABLE SAFETY ANALYSES

The design basis of the TSV Closure function is established by the analysis for the main steam line break (MSLB) as discussed in the UFSAR, Section 15.13 (Ref. 1). TSV closure is necessary to stop steam flow to the turbine (to prevent overcooling) following all reactor trips.

The accident analysis compares several different MSLB events. The MSLB outside containment upstream of the TSV is limiting for offsite dose, although a break in this section of main steam header has a very low probability. The MSLB with ICS low level control and without operator action prior to ten minutes is the limiting case for a post-trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed for include loss of coolant flow events and dropped control rod events and control rod withdrawal events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.2.1, "Regulating Rod Position Limits," LCO 3.2.2, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.3, "QUADRANT POWER TILT (QPT)."

The normal operating band for RCS pressure is between 2125 psig and 2155 psig as measured at the hot leg pressure tap. The safety analyses assume a core exit pressure that is based on the measured pressure and concurrent pressure losses between the two locations. The pressure losses are a function of the loop flow rate, thus different values are allowed for 4 or 3 RCP operation.

Analyses have been performed to establish the pressure, temperature, and flow rate requirements for three pump and four pump operation. These limits are specified in the COLR. The flow limits for three pump operation are substantially lower than for four pump operation. To meet the DNBR criterion, a corresponding maximum power limit is required (see Bases for LCO 3.4.4, "RCS Loops–MODES 1 and 2").

Another set of limits on DNBR related parameters is provided in Safety Limit (SL) 2.1.1, "Reactor Core SLs." Those limits are less restrictive than the limits of LCO 3.4.1, but violation of an SL merits a stricter, more severe Required Action.

The RCS DNB limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

This LCO specifies limits on the monitored process variables: RCS loop (hot leg) pressure, RCS loop average temperature, and RCS total flow rate to ensure that the core operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting DNBR criteria in the event of a DNB limited transient.

The pressure limits are applied to the loop with the highest pressure. The temperature limits are applied to the loop with the lowest loop average temperature for the condition in which there is a 0°F ΔTc setpoint

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 (continued)

restored to a normal operation, steady state condition following load changes and other expected transient operations. The RCS pressure value specified in the COLR is dependent on the number of pumps in operation and has been adjusted to account for the pressure loss difference between the core exit and the measurement location. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation is within safety analysis assumptions. A Note has been added to indicate the pressure limits for three pumps operating is applied to the loop with the highest pressure.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for loop average temperature is sufficient to ensure that the RCS coolant temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions. A Note has been added to indicate the temperature limits for three pumps operating are applied to the loop with the lowest loop average temperature for the condition in which there is a 0°F ΔTc setpoint

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a calorimetric heat balance once every 18 months allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow is greater than or equal to the minimum required RCS flow rate specified in the COLR.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the two spring loaded pressurizer safety valves is to provide RCS overpressure protection. Operating in conjunction with the Reactor Protection System (RPS), two valves are used to ensure that the Safety Limit (SL) of 2750 psig is not exceeded for analyzed transients during operation in MODES 1 and 2. Two safety valves are used for portions of MODE 3. For the remainder of MODE 3, MODE 4, MODE 5, and MODE 6 with the reactor head on, overpressure protection is provided by operating procedures and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III (Ref. 1). The setpoint of the pressurizer code safety valves is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967. The safety valves discharge steam from the pressurizer to a quench tank located in the containment. The discharge flow is indicated by an increase in temperature downstream of the safety valves and by an increase in the quench tank temperature and level.

The required lift pressure is 2500 psig ± 3%. The upper and lower pressure limits are based on the requirements of ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967, which limit the rise in pressure within the vessels which they protect to 10% above the design pressure. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

The consequences of exceeding the ASME pressure limit could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

BASES (continued)

APPLICABLE SAFETY ANALYSES

All accident analyses in the UFSAR that require safety valve actuation assume operation of both pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis is also based on operation of both safety valves and assumes that the valves open at the high range of the setting (2500 psig system design pressure plus 3%). These valves must accommodate pressurizer insurges that could occur during a startup, rod withdrawal, ejected rod, or loss of main feedwater. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at < 15% power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this Specification is required to ensure that the accident analysis and design basis calculations remain valid.

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psig) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions and to comply with ASME Code requirements. The valves will be tested per ASME Section XI requirements and returned to service with as-left setpoints of 2500 psig ± 1%. The upper and lower pressure tolerance limits are based on the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967, which limit the rise in pressure within the vessel which they protect, to 10% above the design pressure. Inoperability of one or both valves could result in exceeding the SL if a transient were to occur.

The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and portions of MODE 3 above the LTOP cut in temperature, OPERABILITY of two valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. Portions of MODE 3 are conservatively included, although the listed accidents may not require both safety valves for protection.

BASES

APPLICABILITY (continued)

The LCO is not applicable in MODE 3 when any RCS cold leg temperature is ≤ 325°F, MODE 4 and MODE 5 because LTOP protection is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head detensioned.

The Note allows entry into MODE 3 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is based on an 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this time frame.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the RCPB.

B.1 and B.2

If the Required Action cannot be met within the required Completion Time or if both pressurizer safety valves are inoperable, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to MODE 3 with any RCS cold leg temperature ≤ 325°F within 18 hours. The 12 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. Similarly, the 18 hours allowed is reasonable, based on operating experience, to reach MODE 3 with any RCS cold leg temperature ≤ 325°F without challenging unit systems. With any RCS cold leg temperature at or below 325°F, overpressure protection is provided by LTOP. Reducing the RCS temperature to ≤ 325°F reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 2), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valves setpoint is ± 3% for OPERABILITY; however, the valves are reset to ±1% during the Surveillance to allow for drift. These values include instrument uncertainties.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. ASME, Boiler and Pressure Vessel Code, Section XI.
3. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.2 Turbine Stop Valves (TSVs)

BASES

BACKGROUND

The TSVs partially isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). TSV closure partially terminates flow from the unaffected (intact) steam generator.

Two TSVs are provided for each main steam line and are located outside of containment. The TSVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent the MSSVs and EFW pump's steam supply from being isolated from the steam generators by TSV closure. Closing the TSVs partially isolates each steam generator from the other, and isolates the turbine from the steam generators.

TSV Closure is initiated by a reactor trip. To keep from rapidly cooling down the primary plant by drawing off too much steam, the turbine is tripped when the reactor trips. Two independent and redundant "Reactor Trip Confirmed" signals in the form of contact closures from the control rod drive system will energize two independent turbine trip mechanisms. The Channel A trip circuit will close all four TSVs within a maximum of 1 second. The Channel B trip circuit will close the TSVs within a maximum of 15 seconds.

A discussion of the TSV's function is found in the UFSAR, Section 10.3 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the TSVs is established by the analysis for the main steam line break (MSLB) as discussed in the UFSAR, Section 15.13 (Ref. 2). TSV closure is necessary to stop steam flow to the turbine (to prevent overcooling) following all reactor trips. Another failure considered is the loss of one switchgear.

The accident analysis compares several different MSLB events. The main SLB outside containment upstream of the TSV is limiting for offsite dose. The MSLB with ICS low level control and no operator action prior to ten minutes is the limiting case for a post-trip return to power. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed.

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valve (ADV) Flow Paths

BASES

BACKGROUND

The ADV flow paths provide a method for cooling the unit to decay heat removal (DHR) entry conditions, should the preferred heat sink via the Turbine Bypass System to the condenser not be available, as discussed in the UFSAR (Ref. 2). This is done in conjunction with the secondary cooling water from the Emergency Feedwater (EFW) System.

The steam generator tube rupture (SGTR) analysis (Ref. 3) credits operator action to depressurize the steam generators by opening each of the ADV flow paths.

For each steam generator, the ADV flow path is comprised of the atmospheric dump block valve bypass (1" bypass), the atmospheric vent valve (a 12" block valve), the atmospheric dump control valve (i.e., throttle valve), and the atmospheric vent block valve (i.e., isolation valve). The throttle valve and the isolation valve are in parallel and are located downstream of the atmospheric vent valve.

The atmospheric vent valve should be opened prior to opening the throttle valve or isolation valve. This is accomplished by first opening the atmospheric dump block valve bypass.

This equalizes the differential pressure across the atmospheric vent valve. Once the atmospheric vent valve is opened, the cool down rate is controlled using the throttle valve. If additional relief capacity is needed, the isolation valve can be opened. The capacity of the throttle or isolation valve exceeds decay heat loads and is sufficient to cool down the plant.

BASES

APPLICABLE SAFETY ANALYSIS

The SGTR analysis credits operator action to depressurize the steam generators by opening both ADV flow paths (i.e., the ADV flow path for each steam generator) within 40 minutes of identifying the ruptured steam generator. Within this 40-minute time period, the operators are only required to open the bypass valve, the block valve, and the throttle valve. However, later in the event, the analysis also assumes that the operators will open the isolation valves in each ADV flow path.

The ADV flow paths satisfy Criterion 3 of 10 CFR 50.36 (Ref.1).

LCO

The ADV flow path for each steam generator is required to be OPERABLE. The failure to meet the LCO can result in the inability to depressurize the steam generators following a SGTR.

An ADV flow path is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and each valve which comprises the ADV flow path is capable of opening and closing.

APPLICABILITY

The ADV flow path for each steam generator is required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal. In MODE 4, steam generators are relied upon for heat removal whenever an RCS loop is required to be OPERABLE or operating to satisfy LCO 3.4.5, "RCS Loops - MODE 4" or available to transfer decay heat to satisfy LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled." The steam generators do not contain a significant amount of energy in MODE 4 when the unit is not relying upon a steam generator for heat transfer, and MODES 5 and 6; therefore, the ADV flow paths are not required to be OPERABLE in these MODES and condition.

BASES

ACTIONS

A.1 and A.2

With one or both of the ADV flow path(s) inoperable, the Unit must be placed in a condition in which the LCO does not apply. To achieve this status, the Unit must be paced in at least MODE 3 within 12 hours, and at least MODE 4 without reliance on a steam generator for heat removal within 24 hours. The Completion Times are reasonable, based on operating experience, to reach the required Unit conditions from full power conditions in an orderly manner and without challenging Unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cool down of the RCS, the valves that comprise the ADV flow path for each steam generator must be able to perform the following functions:

a) the atmospheric dump block valve bypass and the atmospheric vent valve must be capable of being opened and closed; and

b) the atmospheric dump control valve and atmospheric vent block valve must be capable of being opened and throttled through their full range.

This SR ensures that the valves that comprise the ADV flow path for each steam generator are cycled through the full control range at least once per 18 months. Performance of inservice testing or use of an ADV flow path during a unit cool down satisfies this requirement. This surveillance does not require the valves to be tested at pressure. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50.36.
2. UFSAR, Section 10.3.
3. UFSAR, Section 15.9.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Unborated Water Source Isolation Valves

BASES

BACKGROUND	During MODE 6 operations, all isolation valves for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position. The Coolant Storage System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution.
APPLICABLE SAFETY ANALYSES	The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 refueling operations is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources when in MODE 6, a boron dilution event as analyzed in the UFSAR is prevented. The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).
LCO	This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and thus avoid a reduction in SDM.
APPLICABILITY	In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

BASES

APPLICABILITY (continued)

For all other applicable MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated. The boron dilution event is applicable in MODES 1 and 6.

ACTIONS

The ACTIONS table has been modified by a Note that allows separate Condition entry for each unborated water source isolation valve.

A.1

Continuation of CORE ALTERATIONS is contingent upon maintaining the unit in compliance with this LCO. With any valve used to isolate unborated water sources not secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of "immediately" for performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

Condition A has been modified by a Note to require that Required Action A.3 be completed whenever Condition A is entered.

A.2

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valves secured closed. Securing the valves in the closed position ensures that the valves cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. Once actions are initiated, they must be continued until the valves are secured in the closed position.

A.3

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed whenever Condition A is entered to demonstrate that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

These valves are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the fuel transfer canal and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under SR 3.9.1.1. This Surveillance demonstrates that the valves are closed through a system walkdown. The 31 day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.

REFERENCES

1. UFSAR, Section 15.4.1.

2. 10 CFR 50.36.

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION

Pursuant to 10 CFR 50.91, Duke has made the determination that this License Amendment Request involves No Significant Hazards by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

1. <u>Involve a significant increase in the probability or consequences of an accident previously evaluated?</u>

   No. The proposed changes to the Technical Specifications, Bases, Updated Final Safety Analysis Report (UFSAR), and Core Operating Limits Report (COLR) incorporate the accident analyses established in Topical Report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology, Revision 1." On February 1, 1999, Duke submitted Topical Report DPC-NE-3005-P to the NRC for approval. The NRC found DPC-NE-3005-P acceptable as noted in SER dated May 25, 1999.

   The analyzed events are initiated by the failure of specific plant structures, systems or components. These proposed changes do not impact the condition or performance of those structures, systems or components.

   The revised accident analyses in DPC-NE-3005-P demonstrate that the applicable acceptance criteria are met. In addition, the calculations show that the applicable radiological and environmental acceptance criteria continue to be met.

   Based on the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed changes do not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. Where setpoints and operating limits have been revised, the revised accident analyses demonstrate that the applicable acceptance criteria are met. As a result, no new failure modes are being introduced.

Based on the above, the proposed changes do not create the possibility of any new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety?

No. The margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The proposed changes do not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. Where setpoints and operating limits have been revised, the revised accident analyses in DPC-NE-3005-P demonstrate that the applicable acceptance criteria are met.

Based on the above, the proposed changes do not involve a significant reduction in a margin of safety.

Conclusion

Based upon the preceding evaluation, performed pursuant to 10 CFR 50.92, Duke has concluded that the proposed changes to the Oconee Nuclear Station Technical Specifications and Bases will not involve a significant hazards consideration.

ATTACHMENT 5

ENVIRONMENTAL ASSESSMENT

Pursuant to 10 CFR 51.22(b), an evaluation of this LAR is being performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) or 10 CFR 51.22(c)(10) of the regulations.

This LAR for the Oconee Technical Specifications proposes changes to allow implementation of reactor fuel cycle 18 on Unit 2 (O2C18). The reload design for O2C18 was accomplished using DPC-NE-3005-P. The NRC approved the topical report in a Safety Evaluation Report issued on February 25, 1999.

If it can be determined there are:

1) No significant hazards considerations;

2) No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite; and

3) No significant increase in individual or cumulative occupational radiation exposures involved;

then this LAR will qualify for categorical exclusion from the requirement to perform an environmental assessment/impact statement.

For Item 1, as listed above, Attachment 3 of this reload submittal package documents the determination of no significant hazards.

The associated dose analysis show that the O2C18 reload will continue to meet the applicable radiological and environmental acceptance criteria; thus, this LAR will meet the criteria for categorical exclusion from performing an environmental assessment/impact statement.