System Energy

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May 17, 1990

U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D.C. 20555

Attention: Document Control Desk

Gentlemen:

SUBJECT: Grand Gulf Nuclear Station
Unit 1
Docket No. 50-416
License No. NPF-29
Improved Technical Specification
Development Program
AECM-90/0089

System Energy Resources, Inc. (SERI) is currently involved with the General Electric Boiling Water Reactor (BWR) Owners Group as the BWR-6 lead plant for the development of Improved Technical Specifications (ITS). SERI is in the process of developing a plant specific technical specification (PSTS) and subsequent license amendment application based on the BWR Owners' Group Improved BWR Technical Specifications (NEDC-31681). The process involves review of the PSTS by a team of individuals from nuclear and design engineering, licensing and plant operations organizations as well as review by the Plant Safety Review Committee and the Safety Review Committee.

In recent discussions with the NRC-OTSB, SERI was requested to provide to the Staff preliminary drafts of the PSTS in order to facilitate the Staff's validation of the BWR Owners' Group ITS. Pursuant to that request, SERI is providing for your information and preliminary review draft technical specifications for the Power Distribution Limits (3.2), Reactor Coolant System (3.4), Emergency Core Cooling Systems (3.5), and Plant Systems (3.7) prepared under the SERI program for Development of Improved Plant Specific Technical Specifications for Grand Gulf Nuclear Station (GGNS).

Along with each Limiting Condition for Operation (LCO), you will find 1) A Revision Summary Sheet which describes the changes from the current GGNS Technical Specification to the PSTS and 2) A draft bases section for each LCO.

This submittal is made, of course, with the understanding that the drafts provided are only for information at this time and that formal review of the license amendment within SERI has not been completed. Changes, therefore, are likely to occur as the formal application for an amendment is reviewed and certified.

A001

It is our understanding that SERI and the NRC staff will meet the week of June 11, 1990 to discuss the results of the NRC-OTSB review.

Yours truly,

William T. Cottle

WTC:mtc
Attachment

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ATTACHMENT TO AECM-90/0089

SYSTEM ENERGY RESOURCES, INC.
GRAND GULF NUCLEAR STATION

TECHNICAL SPECIFICATIONS IMPROVEMENT PROGRAM

PLANT SPECIFIC TECHNICAL SPECIFICATIONS

CHAPTERS 3.2, 3.4, 3.5 AND 3.7

REVISION SUMMARY SHEET CATEGORY KEY

A. CATEGORIES

1. **ADMINISTRATIVE** - a change which is editorial in nature, involves the movement of requirements within the Technical Specifications without affecting their technical content, simply reformats a requirement, or clarifies the Technical Specification (such as deleting a footnote no longer applicable due to a technical change to a requirement).

2. **RELOCATED** - a change which moves requirements from the Technical Specifications to the Bases, the UFSAR, procedures or other documents.

3A. **TECHNICAL CHANGE, MORE RESTRICTIVE** - a change which adds a requirement to the Technical Specifications or revises an existing requirement to be more stringent.

3B. **TECHNICAL CHANGE, LESS RESTRICTIVE** - a change which revises an existing requirement such that more restoration/completion time is provided or fewer compensatory measures are necessary.

4. **DELETED** - a change which removes requirements from the Technical Specifications without being relocated and without an adequate justification in the BWROG comparison document. Most of the changes in this category are expected to be GGNS-specific requirements which are not in the BWR/6 Standard Technical Specifications. Justification can be provided to support deletion of the requirement or a recommendation made to place the requirement back into the Technical Specifications or to relocate the requirement to another controlled document as discussed in A.2 above.

B. CONVENTIONS

1. A change in which a requirement is moved from an LCO to an LCO other than its associated LCO in the proposed Tech Specs will be included in two LCO review packages (e.g., a requirement moved from LCO 3.1.2 to LCO 3.3.1 will appear in both packages). If the change is **ADMINISTRATIVE**, it will appear as a Category **1** change in both packages. If the change involves a **TECHNICAL CHANGE**, it will appear as a Category **3A** or **3B** change in the LCO package associated with its new location and as a Category **1** change in the LCO package for its previous location. This convention will result in the change only being technically justified one time.

2. References to the existing Tech Specs and the proposed Tech Specs can be distinguished as follows:

    a. **CONDITIONS, REQUIRED ACTIONS** and **COMPLETION TIMES** always refer to the proposed Tech Specs.

    b. **ACTION** or **ACTIONS** always refers to the existing Tech Specs.

    c. **SR 3.x.x** always refers to the proposed Tech Specs while **SR 4.x.x** always refers to the existing Tech Specs.

CHAPTER 3.2

POWER DISTRIBUTION LIMITS

CHAPTER 3.2
POWER DISTRIBUTION LIMITS
TABLE OF CONTENTS

Grand Gulf Nuclear Station
Technical Specification Improvement Program

Revision Summary Sheet

Proposed LCO/Section: 3.2.1 Rev. 1 APLHGR

Item	Change Description	Category
1	LCO 3.2.1 is reformatted from LIMITING CONDITION FOR OPERATION 3.2.1.	1
2	The APLHGR limits are relocated to the Current Cycle Safety Analysis.	2
3	The applicability wording is revised to remove the MODE 1 reference since it is implicitly derived from the power condition.	1
4	CONDITIONS A and B are reformatted from the ACTION statement.	1
5	The 15 minute limit to initiate corrective action specified in the ACTION statement is deleted because the 2 hour limit to restore the parameter within limits is considered to be adequate given the low probability of a transient or accident occurring during this interval.	3B
6	SR 3.2.1.1 is reformatted from SR 4.2.1.a and SR 4.2.1.b except that a surveillance is required once within 12 hours after exceeding 25% RTP instead of at 15% power plateaus.	3B
7	SR 4.2.1.c is deleted. Operation with APLHGR equal to its limit is highly unlikely since margin to the limit is routinely maintained so an increased surveillance frequency is unnecessary.	3B
8	SR 4.2.1.d is deleted based upon the "once within" provision added to SR 3.2.1.1.	1
9	SR 4.2.1.b is deleted. The power increase surveillance has been interpreted differently throughout the industry. The daily surveillance in conjunction with testing within 12 hours after exceeding 25% RTP is considered adequate monitoring.	3B
10	A CROSS REFERENCE is added.	1

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LCO 3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall be less than or equal to the limits specified in the CURRENT CYCLE SAFETY ANALYSIS.

APPLICABILITY: THERMAL POWER ≥ 25% of RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any APLHGR greater than the required limits.	A.1 Restore APLHGR to less than or equal to the required limits.	2 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to < 25% of RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify all APLHGRs are less than or equal to the required limits.	Once within 12 hours after ≥ 25% of RTP AND Once per 24 hours thereafter

CROSS-REFERENCES

TITLE	NUMBER
Recirculation Loops Operating	3.4.1

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

BASES

BACKGROUND

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is a measure of the average linear heat generation rate of all the fuel rods in a fuel assembly at any axial location. Limits on APLHGR are specified to assure that the fuel design limits identified in Reference 1 will not be exceeded during anticipated operational occurrences and that the peak cladding temperature (PCT) during the postulated design basis loss-of-coolant accident (LOCA) will not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents, anticipated operational transients and normal operation that determine the APLHGR limits are presented in References 1, 2, 3 and 4.

Fuel design evaluations are performed to demonstrate that the cladding 1% plastic strain and other fuel design limits described in Reference 1 are not exceeded during anticipated operational occurrences for operation with LINEAR HEAT GENERATION RATES (LHGR's) up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during the limiting anticipated operational occurrences (Ref. 4). Flow dependent APLGHR limits are determined using the three dimensional BWR simulator code (Ref. 5) to analyze slow flow runout transients. The flow-dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. $MAPFAC_f$ curves are based on the maximum credible flow runout transient for Loop Manual and Non-Loop Manual operation. The result of a single failure or single operator error during Loop Manual operation is the runout of only one loop because both recirculation loops are under independent control. Non-Loop Manual operational modes allow simultaneous runout of both loops because a single controller regulates core flow.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power-dependent multipliers ($MAPFAC_p$) are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below that where turbine stop valve closure and turbine control valve fast closure scram trips are bypassed both high and low core flow ~~MAPFACp~~ $MAPFAC_p$ limits are provided for operation at power levels between 25% of RATED THERMAL POWER and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and anticipated operational occurrences. A complete discussion of the analysis is provided in References 1 and 4.

LOCA analyses are performed to ensure the above determined APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models which are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code used in the analysis is provided in Reference 6. The PCT following a postulated LOCA is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. The LOCA analysis was performed at conservatively higher APLHGR values, relative to the requirements of 10CFR50.46, Appendix K.

For single recirculation loop operation, the MAPFAC multiplier is limited to a maximum of 0.86 (Ref. 3). This is due to the conservative analysis assumption of earlier departure from nucleate ~~nuclear~~ boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA.

APLHGR satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 7.

LCO

The APLHGR limits specified in the CURRENT CYCLE SAFETY ANALYSIS ~~CORE OPERATING LIMITS REPORT~~ are the result of the fuel design analysis and design basis accident and transient analysis. For two recirculation loops operating, the limit is determined by multiplying the smaller of the $MAPFAC_f$ and $MAPFAC_p$ factors times the exposure

(continued)

BASES (continued)

LCO (continued)

dependent APLHGR limits. For operation with only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, the limit is determined by multiplying the exposure dependent APLHGR limit times the smaller of either $MAPFAC_f$, $MAPFAC_p$, or 0.86, where 0.86 has been determined by a specific single recirculation loop analysis (Ref. 4).

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations, LOCA and transient analysis that are assumed to occur from high power level conditions. Design calculations (Ref. 4) and operating experience have shown that as power is reduced, margin to required APLHGR limits increases. This trend continues down to low power levels where entry into MODE 2 occurs. When in MODE 2, the Intermediate Range Monitor (IRM) scram function will provide prompt scram initiation during any significant transient thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels less than or equal to 25% of RTP, the reactor will be operating with substantial margin to APLHGR limits and the specification is not required.

ACTIONS

A.1

Should any APLHGR exceed the required limits, an initial condition of the design basis accident and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR's to within the required limits such that the plant will be operating within analyzed conditions and within design limits of the fuel rods.

B.1

If the APLHGR cannot be restored to within the required limits in two hours, it is required to reduce THERMAL POWER to < 25% of RTP. As discussed in the Bases for Applicability, operation below 25% of RTP results in sufficient margin to the required limits.

Completion Times

The Completion Times are based on industry accepted practice and engineering judgement considering the time to reasonably complete the Required Action.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGR's are required to be initially calculated within 12 hours after THERMAL POWER has exceeded 25% of RTP and then daily thereafter. They are compared to the specified limits to assure that the reactor is operating within the assumptions of the safety analysis. The daily requirement for calculating APLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The 12 hour allowance after exceeding 25% of RTP is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. GGNS-1 Current Cycle Safety Analysis.
2. Grand Gulf FSAR, Chapter 4.
3. Grand Gulf FSAR, Chapter 6.
4. Grand Gulf FSAR, Chapter 15 (including Appendices 15C and 15D.
5. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors", Volume 1, June, 1980.
6. XN-NF-80-19(A), "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM ECCS Evaluation Model, "Volume 2, Revision 1, June, 1981.
7. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

Grand Gulf Nuclear Station
Technical Specification Improvement Program

Revision Summary Sheet

Proposed LCO/Section: 3.2.2 Rev. 1 MCPR

Item	Change Description	Category
1	LCO 3.2.2 is reformatted from LIMITING CONDITION FOR OPERATION 3.2.3 except as noted below.	1
2	The MCPR limits are relocated to the Current Cycle Safety Analysis.	2
3	The applicability wording is revised to remove the MODE 1 reference since it is implicitly derived from the power condition.	1
4	CONDITIONS A and B are reformatted from the ACTION statement except as noted below.	1
5	The 15 minute limit to initiate corrective action specified in the ACTION statement is deleted because the 2 hour limit to restore the limit is considered to be adequate given the low probability of a transient or accident occurring during this interval.	3B
6	SR 3.2.2.1 is reformatted from SR 4.2.3.a and SR 4.2.3.b except that a surveillance is required once within 12 hours after exceeding 25% RTP instead of at 25% power plateaus.	3B
7	SR 4.2.3.c is deleted. Operation with MCPR equal to its limit is highly unlikely since margin to the limit is routinely maintained so an increased surveillance frequency is unnecessary.	3B
8	SR 4.2.3.d is deleted based upon the "once within" provision added to SR 3.2.2.1.	1
9	SR 4.2.3.b is deleted since the power increase surveillance has been interpreted differently throughout the industry. The daily surveillance in conjunction with testing within 12 hours after exceeding 25% RTP is considered adequate monitoring.	3B
10	CROSS REFERENCES are added.	1

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO

LCO 3.2.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the MCPR limit specified in the ~~CORE OPERATING LIMITS REPORT.~~ CURRENT CYCLE SAFETY ANALYSIS.

APPLICABILITY: THERMAL POWER ≥ 25% of RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MCPR less than the required limit.	A.1 Restore MCPR to greater than or equal to the required limit.	2 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to < 25% of RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify MCPR is greater than or equal to the required limit.	Once within 12 hours after ≥ 25% of RTP AND Once per 24 hours thereafter

CROSS-REFERENCES

TITLE	NUMBER
Control Rod Scram Times	3.1.3
End of Cycle Recirculation Pump Trip Instrumentation	3.3.4.2
Recirculation Loops Operating	3.4.1

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO

BASES

BACKGROUND

The MINIMUM CRITICAL POWER RATIO (MCPR) is a measure of the operating fuel assembly power relative to the fuel assembly power that would result in the onset of boiling transition. The Safety Limit MCPR is set such that 99.9% of the fuel rods will avoid boiling transition if the limit is not violated (refer to the Bases for LCO 2.1.2). For the purpose of establishing reactor operating limits, damage of the fuel rod cladding is assumed to occur, although fuel damage would not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1). The operating limit MCPR is established to assure that the safety limit is not exceeded during anticipated operational occurrences.

The onset of transition boiling is a phenomena that is readily detected during the testing of various bundle designs. Based on this experimental data, correlations have been developed that are used to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., pressure, mass flux, subcooling, etc.). Since plant operating conditions and bundle power levels are relatively easily monitored and determined, monitoring MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the anticipated operational occurrences to establish the operating limit MCPR are presented in the FSAR, Chapters 4, 6 and 15 and in References 2, 3, 4 and 5. To assure that the Safety Limit MCPR is not exceeded during any moderate frequency transient event, limiting transients have been analyzed to determine the largest reduction in Critical Power Ratio (CPR). The type of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion and coolant temperature decrease. The limiting transient yields the largest ΔCPR. When the largest ΔCPR is added to the Safety Limit MCPR, the required operating limit MCPR is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$ respectively) to ensure adherence to fuel design

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

limits during the worst moderate frequency transient (Ref. 3, 4 and 5). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods, using the three dimensional BWR simulator code (Ref. 6) and the multi-channel thermal hydraulic code (Ref. 7). $MCPR_f$ curves are provided based on the maximum credible flow runout transient for Loop Manual and ~~Nonloop~~ Non-Loop Manual Operations. The result of a single failure or operator error during Loop Manual operation is the runout of one loop because both loops are under independent control. Both loops can runout during Non-Loop ~~Nonloop~~ Manual operation because a single controller regulates core flow.

Power dependent MCPR limits ($MCPR_p$) are determined by the three dimensional BWR simulator code (Ref. 6) and the one-dimensional BWR transient code (Ref. 10). Due to the sensitivity of the transient response to initial core flow levels at power levels below that where the turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, a high and low flow operating limit $MCPR_p$ is provided for operating between 25% of RATED THERMAL POWER (RTP) and the previously mentioned bypass power level.

MCPR satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 9.

LCO

The MCPR operating limits specified in the CURRENT CYCLE SAFETY ANALYSIS ~~CORE OPERATING LIMITS REPORT~~ are the result of the transient analysis. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur from high power level conditions. Below 25% of RTP, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. Surveillance of thermal limits below 25% of RTP is unnecessary due to the large inherent margin that assures that the Safety Limit MCPR will not be exceeded even if a limiting transient should occur. Statistical analyses documented in Reference 8 indicate that the nominal value of initial MCPR expected at 25% of RTP is in excess of 3.0. Studies of the variation of limiting transient behavior have been performed over the range of power/flow conditions. These studies (Ref. 5) encompass the range of key actual plant parameter values important to typically limiting transients.

(continued)

BASES (continued)

APPLICABILITY (continued)

The results of these studies demonstrate that margin is expected between performance and MCPR requirements, and that margins increase as power is reduced to 25% of RTP. This trend is expected to continue to the low power range when entry into MODE 2 occurs. When in MODE 2, the Intermediate Range Monitor (IRM) provides rapid scram initiation for any significant power increase transient which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels less than 25% of RTP, the reactor will be operating with substantial margin to MCPR limits and the specification is not required.

ACTIONS

A.1

Should any MCPR be outside the required limits, an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR's to within the required limits such that the plant will be operating within analyzed conditions.

B.1

If the MCPR cannot be restored to within the required limits in two hours, it is required to reduce THERMAL POWER to < 25% of RTP. As discussed in the Bases for Applicability, operation below 25% of RTP results in sufficient margin to the required limits.

Completion Times

The Completion Times are based on industry accepted practice and engineering judgement considering the time to reasonably complete the Required Action.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

MCPR is required to be initially calculated within 12 hours after THERMAL POWER has exceeded 25% of RTP and then daily thereafter. It is compared to the specified limits to assure that the reactor is operating within the assumptions of the safety analysis. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.2.1

very slow when there have not been significant power or control rod changes. The 12 hour allowance after exceeding 25% of RTP is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NUREG-0562, "Fuel Rod Failure as a Consequence of Departure From Nucleate Boiling or Dryout", June 1979.
2. ~~"GGNS~~ GGNS-1 Current Cycle Safety Analysis~~"~~, ~~(latest version)~~
3. Grand Gulf FSAR, Appendix 15B.
4. Grand Gulf FSAR, Appendix 15C.
5. Grand Gulf FSAR, Appendix 15D.
6. "EXXON Nuclear Methodology for BWRs: Neutronics methods for Design and analysis", XN-NF-80-19 (P)(A), Volume 1, (as supplemented)
7. "EXXON Nuclear Methodology for BWRs: THERMEX, Thermal Limits Methodology Summary Description", XN-NF-80-19(P)(A), Volume 3, Revision 2, January 1987.
8. "BWR/6 Generic Rod Withdrawal Error Analysis", Appendix 15B, General Electric Standard Safety Analysis Report (GESSAR-II).
9. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment", November 1987.
10. "Exxon Nuclear Plant Transient Methodology for BWRs" XN-NF-79-71(P), Revision 2, November 1981.

Grand Gulf Nuclear Station
Technical Specification Improvement Program

Revision Summary Sheet

Proposed LCO/Section: 3.2.3 Rev. 1 LHGR

Item	Change Description	Category
1	LCO 3.2.3 is reformatted from LIMITING CONDITION FOR OPERATION 3.2.4.	1
2	The LHGR limits are relocated to the Current Cycle Safety Analysis or comparable document.	2
3	The applicability wording is revised to remove the MODE 1 reference since it is implicitly derived from the power condition.	1
4	CONDITIONS A and B are reformatted from the ACTION statement.	1
5	The 15 minute limit to initiate corrective action specified in the ACTION statement is deleted because the 2 hour limit to restore the parameter is considered to be adequate given the low probability of a transient or accident occurring during this interval.	3B
6	SR 3.2.3.1 is reformatted from SR 4.2.4.a and SR 4.2.4.b except that a surveillance is required once within 12 hours after exceeding 25% RTP instead of at 15% power plateaus.	3B
7	SR 4.2.4.c is deleted. Operation with LHGR equal to its limit is highly unlikely since margin to the limit is routinely maintained so an increased surveillance frequency is unnecessary.	3B
8	SR 4.2.4.d is deleted based upon the "once within" provision added to SR 3.2.3.1.	1
9	SR 4.2.4.b is deleted. The power increase surveillance has been interpreted differently throughout the industry. The daily surveillance in conjunction with testing within 12 hours after exceeding 25% RTP is considered adequate monitoring.	3B

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE

LCO 3.2.3 The LINEAR HEAT GENERATION RATE (LHGR) shall be less than or equal to the limits specified in the CURRENT CYCLE SAFETY ANALYSIS

APPLICABILITY: THERMAL POWER ≥ 25% of RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any LHGR greater than the required limits.	A.1 Restore LHGR to less than or equal to the required limits.	2 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to < 25% of RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify all LHGRs are less than or equal to the required limits.	Once within 12 hours after ≥ 25% of RTP AND Once per 24 hours thereafter

CROSS-REFERENCES: None

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE

BASES

BACKGROUND

The LINEAR HEAT GENERATION RATE (LHGR) is a measure of the heat generation rate of a fuel rod in a fuel assembly at an axial location. Limits on LHGR are specified to assure that fuel design limits will not be exceeded anywhere in the core during normal operation including anticipated operational occurrences. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to assure that fuel system damage, fuel rod failure or inability to cool the fuel will not occur during the anticipated operating conditions identified in Ref. 1.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating fuel system design are presented in FSAR, Chapter 4 and in (Ref. 1). The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR 20, 50 and 100. The mechanisms which could cause fuel damage during operational transients and which are considered in fuel evaluations are (1) rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet and (2) severe overheating of the fuel rod cladding caused by inadequate cooling. A value of 1% plastic strain of the zircaloy cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 2). The Safety Limit MINIMUM CRITICAL POWER RATIO (MCPR) ensures that fuel damage caused by severe overheating of the fuel rod cladding is avoided and is discussed separately in the Bases for LCO 3.2.2.

Fuel design evaluations have been performed and demonstrate that the 1% plastic strain fuel design limit is not exceeded during continuous operation with LHGR's up to the operating limit specified in the CURRENT CYCLE SAFETY ANALYSIS. The analysis also includes allowances for short term transient operation above the operating limit to account for anticipated operational occurrences including consideration of densification power spiking.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	LHGR satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 3.
LCO	LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause 1% cladding plastic strain. The operating limit to accomplish this objective is specified in the CURRENT CYCLE SAFETY ANALYSIS.
APPLICABILITY	The LHGR limit is derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels less than 25% of RATED THERMAL POWER (RTP), the reactor will be operating with substantial margin to LHGR limits and therefore, the specification is only required when operating at or above 25% of RTP.
ACTIONS	<u>A.1</u> Should any LHGR exceed the required limits, an initial condition of the fuel design analysis will not be met. Therefore, prompt action should be taken to restore the LHGR to within the required limits such that the plant will be operating within analyzed conditions. <u>B.1</u> If the LHGR cannot be restored to within the required limits in two hours, it is required to reduce THERMAL POWER to < 25% of RTP. Operation below 25% of RTP results in sufficient margin to the required limits. <u>Completion Times</u> The Completion Times are based on industry accepted practice and engineering judgement considering the time to reasonably complete the Required Action.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

LHGR is required to be initially calculated within 12 hours after THERMAL POWER has exceeded 25% of RTP and then daily thereafter. It is compared to the specified limits to assure that the reactor is operating within the assumptions of the safety analysis. The daily requirement for calculating LHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The 12 hour allowance after exceeding 25% of RTP is acceptable given the large inherent margin to operating limits at lower power levels.

REFERENCES

1. GGNS-1 Current Cycle Safety Analysis (CCSA)

2. NUREG-0800, Standard Review Plan 4.2, "Fuel System Design", Section II.A.2(g).

3. NEDC-31466, "Technical Specification Screening Criteria Application and Risk Assessment", November 1987.

CHAPTER 3.4

REACTOR COOLANT SYSTEM

CHAPTER 3.4
REACTOR COOLANT SYSTEM
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Proposed LCO/Section: 3.4.1 Rev. 1 Recirc Loops Operating

Item	Change Description	Category
1	LCO 3.4.1 is reformatted from LIMITING CONDITION FOR OPERATIONS 3.4.1.1 and 3.4.1.3.	1
2	The LCO statement is revised to only require two recirculation loops in operation or one loop in operation under specified conditions.	3B
3	DELETED	
4	The SLO loop flow limit and the SLO flow control mode requirement is deleted from LCO 3.4.1.1.b.	4
5	The applicability is revised to eliminate reference to Special Test Exception 3.10.4.	3A
6	Footnote '*' to pages 3/4 4-1 and 3/4 4-3 is deleted (see Item 5 above).	3A
7	CONDITION B is developed from ACTION i of LCO 3.4.1.1 except that ACTION i provided 8 hours and CONDITION B provides 24 hours.	3B
8	CONDITION C is reformated from ACTION a of LCO 3.4.1.1.	1
9	SR 3.4.1.1 is reformatted from LCO 3.4.1.3 and SR 4.4.1.3 except as discussed below.	1
10	ACTION c is deleted.	4
11	ACTION b is deleted.	4
12	ACTION d is deleted.	4
13	ACTION e is deleted.	4
14	ACTION f is deleted.	4
15	SR 4.4.1.1.1 is deleted.	4
16	DELETED (See Items 29 and 30)	
17	SR 4.4.1.1.3 is deleted. (See Item 24)	4
18	SR 4.4.1.1.4 is deleted. (See Item 14)	4
19	SR 4.4.1.1.5 is deleted. (See Item 23)	4

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Proposed LCO/Section: 3.4.1 Rev. 1 Recirc Loops Operating

Item	Change Description	Category
20	SR 4.4.1.1.6 is deleted.	4
21	Figure 3.4.1.1-1 is deleted.	4
22	CROSS REFERENCES are added.	1
23	Action h is deleted.	4
24	Action g is deleted.	4
25	Statement in LCO 3.4.1.1 specifying when operation is/is not permissible (Reference to Figure 3.4.1.1-1) is deleted.	4
26	CONDITION A is developed from ACTION 3.4.1.3.	1
27	CONDITION D is added to address MODE 2 operation.	3B+
28	CONDITION E is developed from ACTION b of LCO 3.4.1.3.	1
29	SR 3.4.1.2 is reformatted from SR 4.4.1.1.2.a.	1
30	SR 3.4.1.3 is reformatted from SR 4.4.1.1.2.b.	1
31	LCO 3.4.1.3 uses the term "rated recirculation flow" as opposed to the PSTS term "rated core flow" in SR 3.4.1.1.	3B

3.4 REACTOR COOLANT SYSTEM

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops shall be in operation,

OR

One recirculation loop may be in operation provided the following limits are made applicable:

A. LCO 3.2.1, APLHGR, Single Loop Operation Limits specified in the ~~CORE OPERATING LIMITS REPORT.~~ CURRENT CYCLE SAFETY ANALYSIS.

B. LCO 3.2.2, MCPR, Single Loop Operation Limits specified in the ~~CORE OPERATING LIMITS REPORT~~. CURRENT CYCLE SAFETY ANALYSIS.

C. LCO 3.3.1.1, RPS Instrumentation, Function 2.b of Table 3.3.1.1-1, Allowable Value is reset for single loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. During two loop operation, recirculation loop jet pump flow mismatch outside requirements.	A.1 Restore the loop jet pump flows to within the specified limits.	2 hours
B. With one recirculation loop not in operation and limits for single loop operation not met.	B.1 Satisfy the single loop operation requirements of the LCO.	24 hours from discovery of loop not in operation
C. No recirculation loops in operation in MODE 1.	C.1 Place the Reactor Mode Switch in the Shutdown position.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. No recirculation loops in operation in MODE 2.	D.1 Restore one loop to operation.	6 hours
E. Required Action and associated Completion Time of Conditions A, B or D not met.	E.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.1.1	Verify recirculation loop jet pump flow mismatch is: A. ≤ 10% of rated core flow when operating at < 70% of rated core flow. B. ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow.	24 hours when both loops are in operation
SR 3.4.1.2	Demonstrate recirculation Flow Control Valve (FCV) fails "as is" on loss of hydraulic pressure at the hydraulic unit.	18 months
SR 3.4.1.3	Demonstrate average rate of FCV movement is: A. ≤ 11% of stroke per second opening. AND B. ≤ 11% of stroke per second closing.	18 months

CROSS-REFERENCES

TITLE	NUMBER
AVERAGE PLANAR LINEAR HEAT GENERATION RATE	3.2.1
MINIMUM CRITICAL POWER RATIO	3.2.2
Reactor Protection System Instrumentation	3.3.1.1
Reactor Coolant System Pressure/Temperature Limits	3.4.8

B 3.4 REACTOR COOLANT SYSTEM

B 3.4.1 Recirculation Loops Operating

BASES

BACKGROUND

The reactor recirculation system is designed to provide a forced coolant flow through the core to remove heat from the fuel. The reactor recirculation system consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. Each external loop contains one motor driven recirculation pump, a flow control valve and associated piping, valves, and instrumentation. The recirculation loops are part of the reactor coolant pressure boundary and are located inside the drywell structure. The jet pumps are reactor vessel internals.

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The drive flow and suction flow are mixed in the jet pump throat section. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core. Each recirculation loop is manually started from the control room. The recirculation flow control valves provide regulation of individual recirculation loop drive flows. The flow in each loop is manually controlled.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The operation of the reactor recirculation system is an initial condition assumed in the design basis Loss of Coolant Accident (LOCA) (Ref. 1). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the accident. The initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump reactor coolant to the vessel almost immediately. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds until the jet pump suction is uncovered (Ref. 1). The analyses assume both loops are operating at the same flow prior to the accident. Because an initial recirculation loop jet pump flow mismatch could affect the transient core flow in the intact loop during pump coastdown, flow mismatch is required to be maintained within specified limits. The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 6) which are analyzed in Chapter 15 of the FSAR.

A plant specific LOCA analysis has been performed for Grand Gulf Unit 1 assuming only one operating recirculation loop. This analysis has demonstrated that in the event of a LOCA caused by a pipe break in the operating recirculation loop, the ECCS response will provide adequate core cooling provided the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) requirements are modified (Ref. 2). The transient analyses of Chapter 15 of the FSAR have also been performed for single recirculation loop operation (Ref. 2) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed. During single recirculation loop operation in MODE 1, modifications to the Reactor Protection System (RPS) Average Power Range Monitor (APRM) instrument setpoints are also required to account for the different response of the reactor and different relationships between recirculation drive flow and reactor core flow.

The APLHGR and MCPR requirements for Grand Gulf Unit 1 also account for the effects of a slow, inadvertent increase in recirculation loop flow to maximum for the two loop as well as the single loop operational conditions.

Recirculation Loops Operating satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 3.

(continued)

BASES (continued)

LCO

Two recirculation loops are required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure during a LOCA, caused by a break of the piping of one recirculation loop, or during a slow runout transient, the assumptions of the applicable analyses are satisfied. With only one recirculation loop in operation, modifications to the required APLHGR limits (LCO 3.2.1), MCPR limits (LCO 3.2.2), and APRM Flow Biased Simulated Thermal Power-High setpoint (LCO 3.3.1.1) may be applied to allow continued operation consistent with the assumptions of Reference 2.

APPLICABILITY

Requirements for operation of the reactor recirculation system are necessary during MODES 1 and 2 since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur. During other conditions, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

A.1

During two loop operation, recirculation loop jet pump flow mismatch limits are in compliance with ECCS/LOCA and the flow runout transient analyses criteria. If the flow mismatch is outside the specified limits, the analyses may no longer be bounding. Therefore, only a limited time is allowed to restore the flow mismatch to within acceptable limits.

B.1

A recirculation loop is considered not to be in operation when the pump in that loop is idle. Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analysis applicable to two loop operations. Therefore, only a limited time is allowed to make the single loop operation limits applicable.

C.1

With no recirculation loops in operation in MODE 1, an immediate reactor shutdown is required. This requirement is a core thermal-hydraulic stability restriction.

(continued)

BASES (continued)

ACTIONS (continued)

D.1

With no recirculation loops in operation in MODE 2, a limited time is allowed to restore one loop to operation.

E.1

With the two loop flow mismatch not restored within the Required Completion Time, the single loop requirements of the LCO not met within the Required Completion Time, or a single loop not restored to operating status within the Required Completion Time, the reactor is required to be in MODE 3. In this condition, the recirculation loops are not required to be operating because of the reduced severity of design basis accidents and minimal dependence on the recirculation loop flow characteristics.

Completion Times

All Completion Times are based on industry accepted practice and engineering judgement considering the number of available systems and the time required to reasonably complete the Required Action.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This surveillance requirement ensures the recirculation loop flows are within the allowable limits for mismatch. At low core flow (i.e. < 70% rated core flow), the MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of boiling transition during a LOCA or a slow flow runout transient is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow. the recirculation loop jet pump flow, as used in this surveillance is the summation of the flows from all of the jet pumps associated with a single recirculation loop. The mismatch is measured in terms of percent of rated core flow. This SR is not required when both loops are not in operating since the mismatch limits are meaningless during single loop or natural circulation operation. Operating experience has demonstrated that a 24 hour frequency for this surveillance is adequate.

SR 3.4.1.2

Loss of Hydraulic Power Unit Pressure, which provides the motive force for the FCVs, causes the FCV to lockup in its last demanded position. This surveillance verifies this function.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.3

This surveillance requirement ensures the overall average rate of FCV movement at all positions is maintained within the analyzed limits.

Surveillance Frequencies

In general, surveillance frequencies are based on industry accepted practice and engineering judgement considering the unit conditions required to perform the test, the ease of performing the test and a likelihood of a change in the system/component status.

REFERENCES

1. Grand Gulf Unit 1 FSAR, Section 6.3.3.7.2.
2. Grand Gulf Unit 1 FSAR, Appendix 15C.
3. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment", November 1987.
4. Grand Gulf Unit 1 FSAR, Section 15.3.2.
5. Grand Gulf Unit 1 FSAR, Section 15.4.5.
6. Grand Gulf Unit 1 FSAR, Section 5.4.1.4.1.

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Revision Summary Sheet

Proposed LCO/Section: 3.4.2 Rev. 1 FCVs

Item	Change Description	Category
1	DELETED	
2	DELETED	
3	DELETED	
4	SR 4.4.1.1.2.a is relocated to SR 3.4.1.2.	2
5	SR 4.4.1.1.2.b is relocated to SR 3.4.1.3.	2

NOTE

PSTS LCO 3.4.2 was deleted per CRS 244. The SRs were incorporated into LCO 3.4.1.

3.4 REACTOR COOLANT SYSTEM

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B 3.4 REACTOR COOLANT SYSTEM

B 3.4.2 Section Deleted

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Revision Summary Sheet

Proposed LCO/Section: 3.4.3 Rev. 1 Jet Pumps

Item	Change Description	Category
1	LCO 3.4.3 is reformatted from LIMITING CONDITION FOR OPERATION 3.4.1.2.	1
2	CONDITION A is reformatted from the ACTION statement.	1
3	SR 3.4.3.1 is reformatted from SR 4.4.1.2.1.	1
4	SR 3.4.3.1 criterion c provides a 20% diffuser to lower plenum check in addition to the 10% jet pump flow check.	3B+
5	SR 4.4.1.2.2 is deleted. SR 3.4.3.1 is a "only required" type surveillance.	3B
6	Footnote '*' to page 3/4 3-2 is deleted. It involved initial values supplied by Startup test program.	4
7	REQUIRED ACTION A.2, to be in MODE 4 in 36 hours, was added.	3A

3.4 REACTOR COOLANT SYSTEM

3.4.3 Jet Pumps

LCO 3.4.3 All jet pumps shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more jet pumps inoperable.	A.1 Be in MODE 3.	12 hours
	AND	
	A.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify at least two of the following criteria (A, B and C) are satisifed for each jet pump in each operating loop: A. Recirculation loop drive flow versus FCV position differs by ≤ 10% from established patterns. B. Recirculation loop drive flow versus total core flow differs by ≤ 10% from established patterns. C. Each jet pump diffuser-to-lower plenum differential pressure differs by ≤ 20% from established patterns. OR Each jet pump flow differs by ≤ 10% from established patterns.	-----NOTE----- Only required when greater than 25% RTP. -------------- 24 hours

CROSS-REFERENCES: None

B 3.4 REACTOR COOLANT SYSTEM

B 3.4.3 Jet Pumps

BASES

BACKGROUND

The reactor recirculation system is described in the Background section of the Bases for LCO 3.4.1.

The jet pumps are part of the reactor recirculation system and are designed to provide forced circulation through the core to remove heat from the fuel. The jet pumps are located in the annular region between the core shroud and the vessel inner wall. Because the jet pump suction elevation is at two-thirds core height, the vessel can be reflooded and coolant level maintained at two-thirds core height even with the complete break of a recirculation loop pipe which is located below the jet pump suction elevation.

Each reactor recirculation loop contains 12 jet pumps. Recirculated coolant passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The drive flow and suction flow are mixed in the jet pump throat section. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Jet pump OPERABILITY is an implicit assumption in the design basis Loss of Coolant Accident (LOCA) analysis evaluated in Reference 1. If a beam holding a jet pump in place fails, jet pump displacement and performance degradation could occur, resulting in a reduction in core flow capacity and a lower core flooding elevation. Jet pump displacement and performance degradation could adversely affect the water level in the core during the reflood phase of a LOCA as well as the assumed blowdown flow. The capability of reflooding the core to two-thirds core height is dependent upon the structural integrity of the jet pumps.

Jet Pumps satisfy the requirements of Selection Criterion 3 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 4.

LCO

The structural failure of any of the jet pumps could cause significant degradation in the ability of the jet pumps to allow reflooding to two-thirds core height during a LOCA. OPERABILITY of all jet pumps is required to ensure operation of the recirculation system will be consistent with the assumptions used in the licensing basis analysis (Ref. 1).

APPLICABILITY

The jet pumps are required to be OPERABLE in MODES 1 and 2 which is consistent with the requirements for operation of the recirculation system (Reference LCO 3.4.1). During MODES 3, 4 and 5 the recirculation system is not required to be in operation and insufficient flow is available to evaluate jet pump operability.

ACTIONS

A.1 and A.2

An inoperable jet pump can increase the blowdown area and reduce the capability of reflooding the core during a design basis LOCA. Therefore with one or more jet pumps inoperable, the reactor is required to be shutdown. The Completion Times allow for a controlled and orderly Shutdown of the reactor.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This surveillance requirement is designed to detect significant degradation in jet pump performance that precedes jet pump failure (Ref. 2 and 3). The jet pump failure of concern is a complete mixer displacement due to jet pump beam failure. Jet pump plugging is also of concern since it adds flow resistance to the recirculation loop thereby affecting coastdown flow to the core. Significant degradation is indicated if more than one of three specified criteria confirms unacceptable deviations from established patterns or relationships. The allowable deviations from the established patterns have been developed based on the variations experienced at plants during normal operation and with jet pump assembly failures (Ref. 2 and 3).

The recirculation flow control valve operating characteristics (loop flow versus flow control valve position) are determined by the flow resistance from the loop suction through the jet pump nozzles. A change in the relationship indicates a flow restriction, loss in pump hydraulic performance, leak, or new flow path between the recirculation pump discharge and jet pump nozzle.

Total core flow can be determined from measurements of the recirculation loop drive flows. Once this relationship has been established, increased or reduced total core flow for the same recirculation loop drive flow may be an indication of failures in one or several jet pumps.

Individual jet pumps in a recirculation loop typically do not have the same flow. The unequal flow is due to the drive flow manifold which does not distribute flow equally to all risers, individual jet pump manufacturing and installation tolerances which cause different jet pump efficiencies, and the resistance the jet pump flow encounters in the lower plenum and vessel annulus. The flow (or jet pump diffuser-to-lower plenum differential pressure) pattern or relationship of one jet pump to the loop average is repeatable. An appreciable change in this relationship is an indication that increased (or reduced) resistance has occurred in one of the jet pumps. This may be seen as an increase in the relative flow for a jet pump that has experienced beam cracks.

This surveillance requirement is not required to be performed when THERMAL POWER is $\leq$ 25% of RATED THERMAL POWER because jet pump noise precludes the collection of repeatable and meaningful data during low flow conditions approaching the threshold response of the associated flow instrumentation.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.3.1 (continued)

Also, this surveillance is not applicable for the jet pumps in a loop not operating becuase there is no drive flow in that loop. Operating experience has demonstrated that a 24 hour frequency for this surveillance is adequate when > 25% of RTP.

REFERENCES

1. Grand Gulf FSAR, Section 6.3.

2. GE Service Information Letter No. 330, "Jet Pump Beam Cracks", June 9, 1980.

3. NUREG/CR3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure", November 1984.

4. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment", November 1987.

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Revision Summary Sheet

Proposed LCO/Section: 3.4.4 Rev. 1 SRVs

Item	Change Description	Category
1	LCO 3.4.4 is reformatted from LIMITING CONDITION FOR OPERATION 3.4.2.1 and 3.4.2.2 except as noted below.	1
2	CONDITION A is reformatted from LCO 3.4.2.1 ACTION a.	1
3	SR 3.4.4.1 is developed from LCO 3.4.2.1.	3B+
4	SR 3.4.4.2 is added to check manual operation of the SRVs.	3A+
5	CROSS REFERENCE to related LCO is added.	1
6	ACTION b is deleted. Suppression pool temperature requirements are specified in LCO 3.6.2.1.	3B
7	The SRV tail-pipe pressure switches (which provide monitor/alarm functions only) in LCO 3.4.2.1 item b, ACTION c and SR 4.4.2.1.1 are deleted (or considered in the OPERABILITY requirements for LCO 3.4.4).	4
8	DELETED	
9	Footnote '*' to page 3/4 4-5 and 3/4 4-7 is relocated to BASES for SR 3.4.4.1.	2
10	Footnote '#' to page 3/4 4-5 is deleted. SRV low-low set is handled by LCO.	3B
11	DELETED	
12	Footnote '*' to page 3/4 4-6 is deleted because the NOTE to SR 3.4.4.2 and LCO 3.0.4 perform the intent of the footnote provision.	1
13	CONDITIONS B and C are reformatted from ACTIONS a and b of LCO 3.4.2.2.	1
14	CONDITION D is reformatted from LCO 3.4.2.1 ACTION d and LCO 3.4.2.2 ACTION c.	1
15	CONDITION E is developed from LCO 3.4.2.1 ACTION d and LCO 3.4.2.2 ACTION c.	3B
16	SR 3.4.4.3 is added to perform a CHANNEL CHECK.	3B+
17	SR 3.4.4.4 is developed from SR 4.4.2.1.2.a and SR 4.4.2.2.1.a. The frequency is reduced to 92 days from 31 days.	3B

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Revision Summary Sheet

Proposed LCO/Section: 3.4.4 Rev. 1 SRVs

Item	Change Description	Category
18	SR 3.4.4.5 is developed from SR 4.4.2.1.2.a and SR 4.4.2.2.1.a. The frequency is reduced to 92 days from 31 days.	3B
19	SR 3.4.4.6 is reformatted from SR 4.4.2.1.2.b and SR 4.4.2.2.1.b.	1
20	SR 3.4.4.7 is developed from SR 4.4.2.1.2.b and SR 4.4.2.2.1.b. A NOTE is added excluding valve actuation from the test.	3B
21	Listing of valves in CTS LCO 3.4.2.2 is relocated to BASES.	2

3.4 REACTOR COOLANT SYSTEM

3.4.4 Safety/Relief Valves

LCO 3.4.4 For the following Safety/Relief Valves (S/RVs):

The safety function of ≥ 7 S/RVs shall be OPERABLE,

AND

The relief function of ≥ 6 additional S/RVs shall be OPERABLE,

AND

The Low-Low Set (LLS) function of 6 S/RVs shall be OPERABLE.

Valve Function	Number of S/RVs	Setpoint (psig)
A. Safety	8	1165 ± 11.6 psi
	6	1180 ± 11.8 psi
	6	1190 ± 11.9 psi
B. Relief	1	1103 ± 15 psi
	10	1113 ± 15 psi
	9	1123 ± 15 psi
C. Low-Low Set		
	1	Open 1033 ± 15 psi Close 926 ± 15 psi
	1	Open 1073 ± 15 psi Close 936 ± 15 psi
	4	Open 1113 ± 15 psi Close 946 ± 15 psi

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITIONS	REQUIRED ACTIONS	COMPLETION TIME
A. One or more of the required S/RVs for the safety or relief function inoperable	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours
B. One LLS S/RV inoperable.	B.1 Restore inoperable valve to OPERABLE status.	14 days from discovery of inoperable valve.
C. More than one LLS S/RV inoperable. OR Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours
D. With either Division 1 or 2 of relief or low-low set actuation instrumentation inoperable.	D.1 Restore inoperable instrumentation to OPERABLE status.	7 days
E. Required Action and associated Completion Time of Condition D not met. OR With both Divisions of relief or low-low set actuation instrumentation inoperable.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Demonstrate the safety function lift setpoints of the required S/RVs.	According to SR 3.0.5 OR 18 months
SR 3.4.4.2	Demonstrate each required S/RV opens when manually actuated.	-----NOTE----- Only required within 12 hours when reactor steam dome pressure is adequate to perform the test. -------------- 18 months
SR 3.4.4.3	Perform a CHANNEL CHECK.	12 hours
SR 3.4.4.4	Calibrate the trip unit.	92 days
SR 3.4.4.5	Perform a CHANNEL FUNCTION TEST.	92 days
SR 3.4.4.6	Perform a CHANNEL CALIBRATION.	18 months
SR 3.4.4.7	--------------Note-------------- Valve actuation may be excluded. ------------------------------ Perform a LOGIC SYSTEM FUNCTION TEST.	18 months

CROSS REFERENCES

TITLE	NUMBER
ECCS - Operating	3.5.1

B 3.4 REACTOR COOLANT SYSTEM

B 3.4.4 Safety/Relief Valves

BASES

BACKGROUND

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions. As part of the nuclear pressure relief system, the size and number of safety/relief valves (S/RVs) are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary.

The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool. The S/RVs can actuate by either of two modes - the safety mode or the relief mode.

In the safety mode (or spring mode of operation), the direct action of the steam pressure in the main steam lines will act against a spring-loaded disk that will pop open when the valve inlet pressure exceeds the spring force.

For the relief mode of operation, each SRV has two pressure actuation trip systems, Division 1 and Division 2. The Division 1 trip system consists of an "A" and an "E" channel of pressure instrumentation. The Division 2 trip system consists of a "B" and an "F" channel of pressure instrumentation. Each channel consists of a separate trip unit and associated logic such that when RPV pressure (as sensed by a pressure transmitter) reaches the set point of the trip unit, the associated relay logic energizes and provides a permissive to energize the respective divisional solenoid for the appropriate SRV. If the other channel in that trip system is then placed in the trip state the solenoid operated air valve will open (solenoid energized) and allow air to port to the pneumatic operator for the SRV, and the SRV will open. Eight of the S/RVs that provide the relief function are part of the Automatic Depressurization System (ADS) specified in LCO 3.5.1.

Six of the S/RVs provide the Low-Low Set (LLS) relief function. To ensure that no more than one relief valve reopens following a reactor isolation event, two valves are provided with lower opening and closing set points and four valves are provided with lower closing setpoints only. These setpoints override the normal relief mode setpoints following the initial opening of any of the relief valves and act to hold open these valves

(continued)

BACKGROUND (continued)

longer, thus preventing subsequent reopening of more than one valve. The low-low set mode of operation is activated anytime the relief mode of operation is activated. Relay logic (two logic channels per division) seals in upon initial relief mode activation, providing a permissive to energize the associated division's low-low set SRV solenoids. A subsequent trip (vessel pressure at or above) of the low-low set SRV's associated pressure instrumentation will then complete the logic, energize the SRV's solenoid and open the SRV. The low-low set instrumentation consists of a single pressure channel per trip system for the two valves with the lowest set point while the four valves with the highest low-low set setting have two channels of instrumentation per trip system.

APPLICABLE SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressurization transient in order to prevent the reactor coolant system pressure from reaching the transient pressure safety limit. Evaluations have determined that the most severe transient is the closure of all main steam line isolation valves (MSIVs) followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 1). For the purposes of the analyses, six of the S/RVs are assumed to operate in the relief mode, and seven of the S/RVs in the safety mode. The analysis results indicate the design S/RV capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). Reference 2 discusses additional events which are expected to actuate the S/RVs. From an overpressure standpoint, these events are bounded by the MSIV closure with flux scram event described above.

The LLS relief mode is designed to protect the containment from excessive loads by ensuring that no more than one relief valve reopens subsequent to the first full blowdown on an isolation event. The LLS function minimizes the induced loading on the containment/suppression pool boundary as a result of subsequent S/RV discharge following the first full blowdown on an isolation event. To remain consistent with the containment loads analyses, the LLS mode of the S/RVs must function to ensure subsequent actuation of only one SRV.

The relief instrumentation must function for the S/RVs to function to prevent over-pressurization of the nuclear system and satisfy the requirements of the transient analyses presented in Reference 2.

S/RVs satisfy the requirements of Selection Criterion 3 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 3.

(continued)

BASES (continued)

LCO

Seven S/RVs are required to be OPERABLE in the safety mode, and an additional 6 S/RVs (other than the 7 S/RVs that satisfy the safety function) must be OPERABLE in the relief mode. In Reference 1, an evaluation was performed to establish the parametric relationship between the peak vessel pressure and the number of OPERABLE S/RVs. The results show that with a minimum of 7 S/RVs in the safety mode and 6 S/RVs in the relief mode OPERABLE, the ASME Code limit of 1375 psig is not exceeded.

The S/RV setpoints are established to ensure the ASME Code limit on peak reactor pressure is satisfied. The ASME code specifications require the lowest safety valve be set at or below vessel design pressure (1250 psig) and the highest safety valve be set so the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The transient evaluations in Reference 2 are based on these setpoints, but also include additional uncertainties to account for potential setpoint drift.

~~These~~ six S/RVs with the LLS function must be OPERABLE in the relief mode to satisfy the assumptions of the safety analysis (Ref. 1). The six valves that satisfy the LCO, along with their setpoints are listed below. The requirements of the LCO are applicable to the mechanical and electrical/pneumatic capability of each valve, and its associated instrumentation, to perform its LLS function.

Valve No.	Setpoint Open	Setpoint Close
F051D	1033	926
F051B	1073	936
F047D	1113	946
F047G	1113	946
F051A	1113	946
F051F	1113	946

Operation with less valves OPERABLE than specified, or with setpoints greater than specified, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.

(continued)

BASES (continued)

APPLICABILITY

The specified number of S/RVs and required S/RV and LLS pressure actuation instrumentation must be OPERABLE in MODES 1, 2 and 3 since there is considerable energy in the reactor core and the limiting design basis transients are assumed to occur. The S/RVs may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) system is capable of dissipating the heat being generated. In MODE 4, decay heat levels are low enough such that the RHR system is adequate, and reactor pressure levels are low enough such that the overpressure limit cannot be challenged by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and there is no reactor coolant pressure boundary (RCPB). The S/RV and LLS functions are not needed during these conditions.

ACTIONS

A.1, A.2

With any of the required S/RVs inoperable, the reactor must be in MODE 3 in 12 hours and in MODE 4 in 36 hours. With less than the minimum number of S/RVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. It is therefore necessary for the plant to be in a condition where the S/RVs are not required. The Completion Times allow for a controlled and orderly Shutdown of the reactor.

B.1, C.1, C.2

With one of the 6 LLS S/RVs inoperable, the remaining OPERABLE LLS S/RVs are adequate to perform the designed function. However, the overall reliability is reduced. A limited out of service time (14 days) is therefore allowed to restore the valve to OPERABLE status.

With more than one LLS S/RV inoperable or Required Action B.1 and associated Completion Time not met, a single failure can cause loss of designed function. The reactor is required to be in MODE 3 in 12 hours and in MODE 4 in 36 hours such that the reactor is in a condition where the LLS S/RVs are not required.

D.1

With one division of relief or LLS actuation instrumentation inoperable, the capability still exists through the remaining division for the required number of S/RVs to function in the

(continued)

BASES (continued)

ACTIONS (continued)

relief and/or LLS mode. An allowable out-of-service time is permitted to allow restoration of the inoperable Trip System, based on the reliability and redundancy of the remaining system.

E.1, E.2

With both divisions of relief or LLS actuation instrumentation inoperable or one division inoperable for more than 7 days, the plant must be placed in a condition where the relief and LLS functions are not required.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This surveillance requirement demonstrates that the S/RVs will open at the pressures assumed in the safety analysis of Reference 1. The demonstration of the S/RV lift settings must be performed during shutdown and in accordance with the provisions of SR 3.0.5. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

SR 3.4.4.2

A manual actuation of each relief S/RV, LLS S/RV and safety S/RV is performed to verify the valve is mechanically functioning properly, the solenoids (for the relief and LLS S/RVs) are functioning properly and no blockage exists in the valve discharge line. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Sufficient time is therefore allowed after the required pressure is achieved to perform this test. Adequate pressure at which this test is to be performed is specified in plant procedures. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements, prior to valve installation. If the valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the S/RV can still be considered OPERABLE for the safety function.

SR 3.4.4.3

The performance of a CHANNEL CHECK is the comparison of the indicated parameter values for each of the required OPERABLE channels for a Function. It is based on the assumption that all channel indications should be displaying approximately the

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.4.3 (continued)

same value consistent with expected values for current plant conditions. Consistency is determined by the plant staff and may be based on a combination of the channel instrument uncertainties, indication capability and readability. Comparison with other independent instrument channels measuring the same parameter may also be used for the CHANNEL CHECK. If a channel is outside of the criteria, it may be an indication the instrument has drifted outside of its limit or is not functioning.

SR 3.4.4.4

Calibration of trip units provides a check of the trip setpoints. If during trip unit calibration the associated trip setting is discovered to be less conservative than the Allowable Value, the channel must be declared inoperable. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

SR 3.4.4.5

Performance of a CHANNEL FUNCTIONAL TEST demonstrates the associated channel will function properly when a signal is injected into the logic indicative of a required trip. If during the CHANNEL FUNCTIONAL TEST, the associated trip setting is discovered to be less conservative than the Allowable Value specified, the channel must be declared inoperable.

SR 3.4.4.6

Performance of a CHANNEL CALIBRATION provides a complete check of the channel including the sensor and trip unit. If during the CHANNEL CALIBRATION, the associated trip setting is discovered to be less conservative than the Allowable Value, the channel must be declared inoperable.

SR 3.4.4.7

Performance of a LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The LOGIC SYSTEM FUNCTIONAL TEST tests all logic components, i.e., all relays and contacts, all trip units,

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.4.7 (continued)

solid state logic elements, etc., of a logic circuit, from sensor up to the actuated device. The system functional test of LCOs 3.4.4 and 3.5.1 for S/RVs overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the associated safety function.

Surveillance Frequencies

In general, surveillance frequencies are based on industry accepted practice and engineering judgement considering the unit conditions required to perform the test, the ease of performing the test and a likelihood of a change in the system/component status.

REFERENCES

1. Grand Gulf FSAR, Section 5.2.2.
2. Current Cycle Safety Analysis.
3. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment", November 1987.

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Revision Summary Sheet

Proposed LCO/Section: 3.4.5 Rev. 1 Operational Leakage

Item	Change Description	Category
1	LCO 3.4.5 is reformatted from LIMITING CONDITION FOR OPERATION 3.4.3.2.	1
2	DELETED (Ref. CRS 174)	
3	LCO 3.4.3.2 item d is moved to LCO 3.6.1.6.	1
4	A NOTE is added stating that CONDITIONS A through C can be concurrently entered.	1
5	CONDITION A is reformatted from ACTION b except the shutdown requirements are in CONDITION C (see Item 7).	1
6	CONDITION B is reformatted from ACTION e except the shutdown requirements are in CONDITION C (see Item 7).	1
7	CONDITION C is reformatted from ACTION a and the shutdown requirements from ACTIONS b and e.	1
8	SR 3.4.5.1 provides monitoring leakage on a 12 hour frequency. SR 4.4.3.2.1 provided frequencies ranging from 4 hours to 24 hours.	3B
9	ACTION c is moved to LCO 3.6.1.6.	1
10	ACTION d is deleted. Justification in comparison document addressed only the monitors and not the interlocks.	4
11	The specific leakage detection methods are deleted from SR 4.4.3.2.1. Leak Detection System requirements are provided by LCO 3.3.4.1 and the surveillance frequency specified by SR 3.4.5.1.	3B
12	SR 4.4.3.2.2 is moved to LCO 3.6.1.6.	1
13	SR 4.4.3.2.3 is deleted. Justification in comparison document addressed only the monitors and not the interlocks (see Item 10).	4
14	Table 3.4.3.2-2 is deleted. This change is considered administrative because it addresses an action statement which has been deleted (see Item 10).	1

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Proposed LCO/Section: 3.4.5 Rev. 1 Operational Leakage

Item	Change Description	Category
15	LCO item D is made applicable in MODE 1 only.	3B
16	Table 3.4.3.2-1 is removed from the Tech Specs and relocated based upon guidance in the proposed generic letter.	2
17	Table 3.4.3.2-3 is deleted from Tech Specs and not relocated (see Item 10).	4

3.4 REACTOR COOLANT SYSTEM

3.4.5 Operational Leakage

LCO 3.4.5 Reactor coolant system LEAKAGE shall be limited to:

A. No Pressure Boundary Leakage,

AND

B. ≤ 5 gpm total Unidentified Leakage,

AND

C. ≤ 30 gpm Total Leakage averaged over any 24 hour period,

AND

D. ≤ 2 gpm increase in Unidentified Leakage within any 4 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

------------------------NOTE------------------------
Conditions A through C may be concurrently applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Unidentified Leakage > 5 gpm. OR Total Leakage > 30 gpm. OR Unidentified Leakage > 5 gpm and Total Leakage > 30 gpm.	A.1 Reduce leakage to within the limits.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Unidentified Leakage increase > 2 gpm in any 4 hour period.	B.1 Verify source of leakage increase is not service sensitive Type 304 or 316 austenitic stainless steel.	4 hours
C. Required Actions and associated Completion Times of Condition A or B not met. OR Any Pressure Boundary Leakage.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify the reactor coolant system LEAKAGE is less than or equal to the required limits.	12 hours

CROSS-REFERENCES: None

B 3.4 REACTOR COOLANT SYSTEM

B 3.4.5 Operational Leakage

BASES

BACKGROUND

The reactor coolant system includes systems and components that contain or transport fluids to or from the reactor core. These systems form a major portion of the reactor coolant pressure boundary. The pressure containing components of the reactor coolant system, including the portions of the system out to and including isolation valves, are defined as the Reactor Coolant Pressure Boundary (RCPB). Limits on leakage from the RCPB are required to ensure appropriate action can be taken before the integrity of the nuclear system process barrier is impaired.

The safety significance of leaks from the RCPB can vary widely depending on the source of the leak as well as the leakage rate and duration. Therefore, detection of leakage in the drywell is necessary. Identified Leakage is defined as the leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered and piped to a sump or collecting tank. Also, leakage into the drywell atmosphere from sources that are specifically located and known not to interfere with the operation of Unidentified Leakage detection or not to be a flaw in the RCPB are considered Identified Leakage. Unidentified Leakage is collected in the drywell floor drain sump. Methods for separating the Indentified Leakage from the Unidentified Leakage are necessary to provide prompt and quantitative information to the operators to permit them to take corrective action.

A limited amount of leakage is expected from auxiliary systems within the drywell that cannot be made 100% leaktight. If leakage occurs from these paths, it should be detectable and isolated from the drywell atmosphere if possible, so as not to mask any potentially serious leak should it occur.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of pipe cracks. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage were also considered. The evidence obtained from experiments suggests for leakage somewhat greater than specified for Unidentified Leakage, the probability is small the imperfection or crack associated with such leakage would grow rapidly. The Unidentified Leakage rate limit is established at 5 gpm to allow time for corrective action before the process barrier could be significantly compromised. This limit is a small fraction of the calculated flow from a critical crack in the primary system piping. Based on crack behavior from experimental programs (Ref. 2 and 3) it is estimated that leak rates of hundreds of gpm will precede crack instability (Ref. 4).

There are no applicable safety analyses that assume the Total Leakage limit. The Total Leakage limit is specified based on consideration of inventory makeup capability and sump capacities.

Operational Leakage satisfies the requirements of Selection Criterion 1 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 5.

LCO

No Pressure Boundary Leakage is allowed since the potential exists for a break in the RCPB and a loss of substantial inventory.

The Total Leakage rate consists of all leakage, identified and unidentified, that flows to the drywell floor drain and equipment drain sumps. The Unidentified Leakage rate limit is established to allow time for corrective action before the reactor coolant pressure boundary is significantly compromised and based on a crack size large enough to propagate rapidly.

(continued)

BASES (continued)

LCO (continued)

In addition to the absolute limits discussed above, a limit is placed on the increase in Unidentified Leakage over a specified period. An increase of 2 gpm in any 4 hour period in Unidentified Leakage is indicative of a potential flaw in the RCPB and it should be promptly evaluated to determine the source and extent of the increased leakage. The 2 gpm increase is measured relative to the steady state Leakage value. This allows for temporary changes in leakage that are the expected result of transient conditions (e.g., startup). As such, the 2 gpm increase limit is only applicable in MODE 1 where operating pressures and temperatures have been established.

APPLICABILITY

The potential for RCPB leakage is greatest when the reactor is pressurized. Under these conditions, high stresses are applied to the system piping resulting in the potential for crack growth and possible failure of the RCPB. Therefore, detection and measurement of RCPB leakage is required during MODES 1, 2 and 3. In MODES 4 and 5, operational leakage limits are not required, since the reactor is not pressurized and the potential for leakage and possible pressure boundary failure is reduced.

ACTIONS

A.1

With either the Total Leakage, Unidentified Leakage, or both greater than the required limits, actions should be taken to identify the source of the leak and determine the significance. Because the leakage limits are conservatively below the leakage that would constitute a critical crack size, a limited time is allowed to evaluate the situation. If a change in Unidentified Leakage has been adequately identified and quantified, it may be reclassified and considered as Identified Leakage. However, the Total Leakage limit would remain unchanged.

(continued)

BASES (continued)

ACTIONS (continued)

B.1

An increase of 2 gpm in Unidentified Leakage in a 4 hour period is an indication of a potential flaw in the RCPB and should be promptly evaluated. Although the increase does not necessarily violate the absolute Unidentified Leakage limits, certain susceptible components should be determined to not be the source of the leaks. Reactor coolant system service sensitive Type 304 and 316 austenitic stainless steel piping should be evaluated and eliminated as a potential source of the increased leakage. These components are especially susceptible to intergranular stress corrosion cracking.

C.1, C.2

If the LEAKAGE cannot be restored to within the required limits, the reactor should be in MODE 3 and subsequently in MODE 4. If Pressure Boundary Leakage occurs there is the potential that the flaw in the RCPB could eventually result in a pipe break or other LOCA. Since the area being monitored is inaccessible, the reactor must be in MODE 3 and subsequently in MODE 4 to allow a visual inspection to determine the source of the leak.

Completion Times

All Completion Times are based on industry accepted practice and engineering judgement considering the number of available systems and the time required to reasonably complete the Required Action.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

The reactor coolant system LEAKAGE is monitored by a variety of systems designed to provide alarms when leakage is indicated and to quantify the various types of leakage. Leakage detection is discussed in more detail in the Bases for LCO 3.3.4.1. Changes in sump levels and measured flow rates are monitored to determine actual leakage rates. However, additional methods may be used which quantify leakage within the guidelines of Reference 1. Operating experience has demonstrated that a 12 hour frequency for this surveillance is adequate.

(continued)

BASES (continued)

REFERENCES

1. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

2. GEAP-5620, "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws," April 1968.

3. NUREG-76/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975.

4. Grand Gulf FSAR, Section 5.2.5.5.3.

5. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment", November 1987.

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Revision Summary Sheet

Proposed LCO/Section: 3.4.6 Rev. 1 Specific Activity

Item	Change Description	Category
1	LCO 3.4.6 is reformatted from LIMITING CONDITION FOR OPERATION 3.4.5.	1
2	The 100/E microcuries per gram criterion of LCO 3.4.5 is deleted because iodine monitoring is considered most limiting.	3B
3	The applicability in MODE 4 is deleted because no pressure or steam exists to provide a force or medium to transport activity beyond the vessel.	3B
4	The applicability in MODES 2 and 3 is limited to when any main steam line is not isolated because activity cannot escape a breach outside containment with the MSIVs closed.	3B
5	CONDITION A is reformatted from ACTIONS a.1 and b except as modified per Items 6 and 7.	1
6	REQUIRED ACTIONS A.1 and B.1 are reformatted from ACTION b and Table 4.4.5-1 item 4(a).	1
7	REQUIRED ACTIONS A.2 and B.2 are reformatted from ACTION a.1 except the requirement for HOT SHUTDOWN is removed.	3B
8	SR 3.4.6.1 is reformatted from Table 4.4.5-1 item 2.	1
9	ACTION a.2 is deleted (see Item 2).	3B
10	ACTION b is revised to delete reference to 100/E microcuries per gram (see Item 2).	3B
11	ACTION c is deleted.	4
12	Footnote '*' to page 3/4 4-17 is deleted. It applied to the startup test program and is no longer applicable.	1
13	Table 4.4.5-1 items 1 and 5 are relocated.	2
14	Table 4.4.5-1 item 3 is deleted (see Item 2).	3B
15	Table 4.4.5-1 item 4(b) is deleted.	4
16	Footnote '*' to page 3/4 4-18 is deleted (see Item 2).	3B

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Revision Summary Sheet

Proposed LCO/Section: 3.4.6 Rev. 1 Specific Activity

Item	Change Description	Category
17	Footnote '#' to page 3/4 4-18 is deleted. This provision is addressed by REQUIRED ACTIONS A.1 and B.1 (as reformatted).	1
18	Table 4.4.5-1 is deleted.	4

3.4 REACTOR COOLANT SYSTEM

3.4.6 Specific Activity

LCO 3.4.6 The specific activity of the primary coolant shall be ≤ 0.2 microcuries per gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Primary coolant specific activity > 0.2 but ≤ 4.0 μCi per gram DOSE EQUIVALENT I-131.	A.1 Perform an isotopic analysis for Iodine.	Once per 4 hours
	AND	
	A.2 Restore specific activity to within limits.	48 hours
B. Required Actions and associated Completion Times of Condition A not met.	B.1 Perform an isotopic analysis for Iodine.	Once per 4 hours
OR	AND	
Primary coolant specific activity > 4.0 μCi per gram DOSE EQUIVALENT I-131.	B.2 Isolate all main steam lines.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.6.1	-----------------NOTE----------------- Only required in MODE 1. ------------------------------------- Demonstrate specific activity of primary coolant is ≤ 0.2 μCi per gram DOSE EQUIVALENT I-131.	31 days

CROSS-REFERENCES: None

B 3.4 REACTOR COOLANT SYSTEM

B 3.4.6 Specific Activity

BASES

BACKGROUND	During circulation, the reactor coolant acquires radioactive material due to release of fission products into the coolant and activation of crud particles in the reactor coolant. These radioactive materials in the reactor coolant could contribute to release of radioactive materials into the environment during design basis accidents. Limits on the maximum allowable level of radioactivity in the reactor coolant are established to assure, in the event of a release of any radioactive material to the environment during a design basis accident, radiation doses are maintained within the limits of 10 CFR 100.
APPLICABLE SAFETY ANALYSES	Analytical methods and assumptions involving radioactive material in the primary coolant are presented in FSAR Chapter 15, Accident Analyses. The specific activity in the reactor coolant (source term) is an initial condition assumed for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside the containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is postulated to be terminated by complete closure of the main steam line isolation valves (MSIVs). This release forms the basis for determining off-site doses (Ref. 1). The limitations on the specific activity of the primary coolant ensure the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed 10% of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as site boundary location and meteorological conditions, were not considered in this evaluation. Specific Activity satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 2.

(continued)

BASES (continued)

LCO

The primary coolant specific activity level of ≤ 0.2 microcuries per gram DOSE EQUIVALENT I-131 is required to ensure the source term assumed in the safety analysis of the MSLB is not exceeded such that any release of radioactive material to the environment does not exceed 10 CFR 100 limits.

APPLICABILITY

Limitations on levels of primary coolant radioactivity are applicable during MODE 1 and MODES 2 and 3 with any main steam line not isolated since there is an escape path for release of radioactive material from the coolant to the environment in the event of a MSLB outside of the primary containment. During MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

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

A primary coolant specific activity level > 0.2 μCi per gram DOSE EQUIVALENT I-131 indicates the presence of some abnormality in plant operations. The range between 0.2 μCi and 4.0 μCi is acceptable for up to 48 continuous hours to account for potential iodine spiking that may occur following changes in THERMAL POWER. Increased surveillance of the reactor coolant specific activity during this period is required to closely monitor the condition and determine if additional limits are exceeded.

If coolant specific activity cannot be restored ≤ 0.2 μCi within 48 hours, or when coolant activity is > 4.0 μCi, the main steam lines are required to be isolated. This action precludes the possibility of release of radioactive material to the environment in excess of the requirements of 10 CFR 100, during a postulated MSLB accident.

Completion Times

All Completion Times are based on industry accepted practice and engineering judgement considering the number of available systems and the time required to reasonably complete the Required Action.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

Isotopic analysis for DOSE EQUIVALENT I-131 concentration is necessary to determine that the specified maximum primary coolant activity is $\leq$ 0.2 microcuries per gram DOSE EQUIVALENT I-131 during steady state operation. Operating experience has demonstrated that a 31 day frequency for this surveillance is adequate.

Surveillance Frequencies

In general, surveillance frequencies are based on industry accepted practice and engineering judgement considering the unit conditions required to perform the test, the ease of performing the test and a likelihood of a change in the system/component status.

REFERENCES

1. Grand Gulf, FSAR Section 15.6.4.

2. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment", November 1987.

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Revision Summary Sheet

Proposed LCO/Section: 3.4.7 Rev. 1 RHR-Shutdown

Item	Change Description	Category
1	LCO 3.4.7 is reformatted from LIMITING CONDITIONS FOR OPERATION 3.4.9.1 and 3.4.9.2.	1
2	The LCO statement is revised to only require that two RHR shutdown cooling subsystems be OPERABLE. The requirement for at least one RHR SDC loop to be in operation or one recirculation pump to be running is deleted.	3B
3	The reference to one RHR pump and heat exchanger in the LCO statement for LCOs 3.4.9.1 and 3.4.9.2 is relocated to the BASES.	2
4	The applicability in MODE 4 is revised to a condition based upon decay heat generation and heat removal loads.	3B
5	CONDITION A is developed from ACTION a for LCO 3.4.9.1.	3B
6	CONDITION B and C are developed from ACTION a for LCOs 3.4.9.1 and 3.4.9.2.	3B
7	The requirement to demonstrate the alternate method of decay heat removal is deleted. Alternate capability car be provided by plant specific administrative controls.	2
8	ACTION b for LCOs 3.4.9.1 and 3.4.9.2 is deleted. Coolant circulation is available without forced circulation and can be administratively controlled.	2
9	SR 3.4.7.1 is added to verify the capability to establish the correct valve alignment for the shutdown cooling mode of RHR.	3A+
10	SRs 4.4.9.1 and 4.4.9.2 are deleted. The revision to the LCO statement (see Item 2) removes the need to periodically verify a SDC loop is operating.	3B

Grand Gulf Nuclear Station
Technical Specification Improvement Program

Revision Summary Sheet

Proposed LCO/Section: 3.4.7 Rev. 1 RHR-Shutdown

Item	Change Description	Category
11	Footnote '#' to pages 3/4 4-26 and 3/4 4-27 is deleted. The revision to the LCO statement (see Item 2) eliminates the need for the other loop to be in operation during this time.	3B
12	Footnote '*' to pages 3/4 4-26 and 3/4 4-27 is deleted. The revision to the LCO statement (see Item 2) removes the requirement for a RHR SDC loop to always be in operation.	3B
13	Footnote '##' to pages 3/4 4-26 and 3/4 4-27 is deleted. The revision to the LCO statement (see Item 2) removes the requirement for a RHR SDC loop to always be in operation.	3B
14	Footnote '**' to page 3/4 4-26 is deleted. This provision is addressed in CONDITION C.	3B
15	Footnote '**' to page 3/4 4-27 is deleted. It applied to RFO3 only.	1
16	Footnote '***' is deleted. It applied to RFO3 only.	1
17	CROSS REFERENCES are added.	1

3.4 REACTOR COOLANT SYSTEM

3.4.7 Residual Heat Removal - Shutdown

LCO 3.4.7 Two Residual Heat Removal (RHR) shutdown cooling subsystems shall be OPERABLE.

APPLICABILITY: MODE 3 with reactor steam dome pressure < the RHR cut-in permissive pressure,
MODE 4 with heat losses to ambient not sufficient to maintain average reactor coolant temperature ≤ 200°F.

ACTIONS

CONDITIONS	REQUIRED ACTION	COMPLETION TIME
A. One of the required RHR shutdown cooling subsystems inoperable.	A.1 Restore the required RHR subsystem to OPERABLE status. OR A.2 Provide an alternate method capable of decay heat removal for the inoperable subsystems.	 As soon as practicable
B. No RHR shutdown cooling subsystem OPERABLE.	B.1 Restore at least one RHR subsystems to OPERABLE status.	As soon as practicable
C. Required Action and associated Completion Time of Condition B not met.	C.1 Provide an alternate method capable of decay heat removal for each inoperable subsystem.	As soon as practicable

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.7.1	Verify for the required RHR shutdown cooling subsystem(s) each manual, power operated, or automatic valve in the flow path, not locked, sealed or otherwise secured in position, is in the correct position or is capable of being manually aligned in the correct position.	31 days OR 24 hours when reactor steam dome pressure is < the RHR cut-in permissive pressure.

CROSS-REFERENCES

TITLE	NUMBER
ECCS - Operating	3.5.1
ECCS - Shutdown	3.5.2
Residual Heat Removal Suppression Pool Cooling	3.6.2.3

B 3.4 REACTOR COOLANT SYSTEM

B 3.4.7 Residual Heat Removal - Shutdown

BASES

BACKGROUND

Irradiated fuel in the reactor pressure vessel (RPV) generates decay heat during normal and abnormal shutdown conditions, potentially resulting in an increase in the temperature of the reactor coolant. The reactor vessel and internals also contain sensible or stored heat energy. This decay and sensible heat is required to be removed such that the reactor coolant temperature can be reduced to or maintained at ≤ 200°F. A system capable of removing decay heat is therefore required to perform this function.

The two shutdown cooling loops of the Residual Heat Removal (RHR) system provide decay heat removal. Each loop consists of a motor driven pump, two heat exchangers in series, and the associated piping and valves. Both loops have a common suction from the same recirculation loop. Each pump discharges the reactor coolant, after it has been cooled by circulation through the respective heat exchangers, to the reactor. The RHR heat exchangers transfer heat to the Standby Service Water System (LCO 3.7.1 and LCO 3.7.2). The RHR shutdown cooling mode is a manually controlled system.

APPLICABLE SAFETY ANALYSES

Residual heat removal through operation of the shutdown cooling mode of the RHR system is not required for mitigation of any postulated transients or accidents evaluated in the safety analyses. However, the NRC Interim Policy Statement (Ref. 1) requires the RHR system be retained in the Technical Specifications even though the shutdown cooling mode of RHR did not satisfy any of the selection criteria (Ref. 2).

(continued)

BASES (continued)

LCO

Two shutdown cooling subsystems are required to be OPERABLE. An OPERABLE RHR shutdown cooling subsystem consists of one RHR pump, two heat exchangers in series, and the associated piping and valves. Additionally, each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODES 3 and 4, one shutdown cooling subsystem of the RHR can provide the required cooling to maintain the desired temperature. Two subsystems are required to be OPERABLE to provide redundancy. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required.

APPLICABILITY

Decay heat removal at reactor pressures above the RHR cut-in permissive pressure is typically accomplished by condensing steam from the RPV in the main condenser. When the reactor pressure is below the RHR cut-in permissive pressure, the RHR system may be operated in the shutdown cooling mode. Operation of the RHR system in the shutdown cooling mode above this pressure is not allowed because the coolant pressure may exceed the design pressure of the shutdown cooling piping. In MODE 3 operation of a subsystem to remove the decay heat may be required to either reduce or maintain coolant temperature. If ambient losses are insufficient continued operation of a shutdown cooling subsystem may be required to reach and maintain reactor coolant temperatures ≤ 200°F, which corresponds to MODE 4. The requirements for decay heat removal in MODE 5 are discussed in LCO 3.9.8 and LCO 3.9.9.

(continued)

BASES (continued)

ACTIONS

A.1, A.2, B.1, C.1

With one RHR shutdown cooling subsystem inoperable for decay heat removal, the remaining OPERABLE subsystem or an alternate method of decay heat removal can provide the necessary decay heat removal. The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as contributing to the alternate method capability. Loss of one RHR shutdown cooling subsystem reduces the overall system reliability, therefore the subsystem should be restored or an alternate method of decay heat removal should be provided as soon as practicable. If one inoperable subsystem cannot be restored to OPERABLE status or with both subsystems inoperable, an alternate method of decay heat removal is required to be made available for each inoperable subsystem to restore cooling capability as soon as practicable.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

Verification that all valves of the required RHR shutdown cooling subsystem are in the correct position ensures the proper flow path. Valves not in the correct position must be capable of manual realignment either from the control room or at the valve location. Verification of this capability is provided by actuation of the valve from the control room or the current inservice inspection reports. Operating experience has demonstrated that a 31 day frequency for this surveillance is adequate. Because some of the required valves are interlocked closed when above the RHR cut-in permissive pressure, an allowance is provided to test the valves within 24 hours after pressure has been reduced below the cut-in permissive pressure. This allows conditions to be established under which the test may be performed.

(continued)

BASES (continued)

REFERENCES

1. 52FR3788, Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

2. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

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Revision Summary Sheet

Proposed LCO/Section: 3.4.8 Rev. 1 P/T Limits

Item	Change Description	Category
1	LCO 3.4.8 is reformatted from LIMITING CONDITION FOR OPERATION 3.4.6.1.	1
2	Figure 3.4.6.1-1 and the RCS temperature and pressure limitations are relocated to the UFSAR.	2
3	The RCS heatup and cooldown rate limitations are relocated to the UFSAR.	2
4	LCO 3.4.6.1 Item c is deleted.	3B
5	LCO 3.4.6.1 item d is relocated.	2
6	CONDITION A is reformatted from the ACTION statement except as modified per Item 7.	1
7	REQUIRED ACTION A.2 has 72 hours specified for its COMPLETION TIME. The time was unstated previously.	3B
8	CONDITION B is reformatted from the ACTION statement.	1
9	SR 3.4.8.1 is reformatted from SR 4.4.6.1.1.	1
10	SR 3.4.8.2 is reformatted from SR 4.4.6.1.1.	1
11	SR 3.4.8.3 is reformatted from LCO 3.4.1.4 and SR 4.4.1.4.	1
12	SR 3.4.8.4 is reformatted from LCO 3.4.1.4 and SR 4.4.1.4.	1
13	SR 3.4.8.5 is reformatted from SR 4.4.6.1.3.	1
14	SR 4.4.6.1.4 is relocated.	2
15	SR 4.4.6.1.5 is deleted. The flux wire specimens were removed during a previous outage.	3B
16	Table 4.4.6.1.3-1 is relocated (see Item 14).	2
17	CROSS REFERENCE is added.	1

3.4 REACTOR COOLANT SYSTEM

3.4.8 Reactor Coolant System Pressure/Temperature Limits

LCO 3.4.8 The Reactor Coolant System (RCS) temperature and reactor vessel pressure shall be maintained within the Pressure/Temperature (PT) Limits.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------NOTE-------- Required Actions A.1 and A.2 must be completed whenever this Condition is entered. ------------------- Operation outside the PT Limits.	A.1 Restore RCS temperature and reactor vessel pressure to within the PT Limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Actions and associated Completion Times of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	------------------NOTE------------------ Only required during system heatup, cooldown and inservice leak and hydrostatic testing. -- Verify RCS pressure and temperature are within the PT Limits.	 30 minutes
SR 3.4.8.2	Verify RCS pressure and temperature are within the PT Limit Curve criticality limit.	Once within 15 minutes prior to initial control rod withdrawal for the purpose of achieving criticality
SR 3.4.8.3	------------------NOTE------------------ Only required in MODES 1,2,3, and 4 with reactor steam dome pressure $\geq$ 25 psig. -- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel coolant temperature is $\leq$ 100°F.	 Once within 15 minutes prior to each startup of a recirculation loop pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.8.4	-------------------NOTE---------------- Only required in MODES 1,2,3, and 4. --------------------------------------- Verify the difference between the reactor coolant temperature within the recirculation loop to be started and the reactor pressure vessel coolant temperature is ≤ 50°F.	 Once within 15 minutes prior to each startup of a recirculation loop pump
SR 3.4.8.5	Verify the reactor vessel flange and head flange temperature are ≥ 70°F.	12 hours when in MODE 4 with reactor coolant system temperature ≤ 100°F __AND__ 30 minutes when in MODE 4 with reactor coolant system temperature ≤ 80°F __AND__ 30 minutes when tensioning the reactor vessel head bolting studs

CROSS-REFERENCES

TITLE	NUMBER
Recirculation Loops Operating	3.4.1
~~Inservice Leak and Hydrostatic Testing Operation~~	~~3.10.1~~

B 3.4 REACTOR COOLANT SYSTEM

B 3.4.8 <u>Reactor Coolant System Pressure/Temperature Limits</u>

BASES

BACKGROUND

All components in the reactor coolant system (RCS) are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

The purpose of this specification is to establish operating limits that provide a conservative margin to brittle failure of major piping and pressure vessel components of the Reactor Coolant Pressure Boundary (RCPB). Of the major components within the RCPB, the reactor vessel (including feedwater nozzles, vessel flanges, shell and closure studs) is the component most subject to brittle failure and therefore the component for which the technical specification limits is most pertinent.

The basis of the pressure and temperature (PT) limits is found in Appendix G to 10 CFR 50 (Ref. 1). Appendix G requires that limits be established, and that the limits be based on specific fracture toughness requirements for RCPB materials such that an adequate margin to brittle failure will be provided during operational occurrences. 10 CFR 50 Appendix G mandates the use of ASME Section III, Appendix G (Ref. 2).

The concern addressed by Appendix G is that undetected flaws could exist in the RCPB components, which if subjected to unusual pressure and/or thermal stresses, could result in non-ductile (brittle) failure. Certain reactor coolant system PT combinations can cause stress concentrations at flaw locations which in turn could cause flaw growth, resulting in failure before the ultimate strength of the material is attained. Flaw growth is resisted by the material toughness.

Toughness is a material property which depends on the alloy microstructure. Toughness of steels vary with ambient temperature, and is lower at room temperature than at reactor operation temperature. Furthermore, toughness is negatively affected by neutron irradiation. The cumulative effect of neutron irradiation (fluence) causes the toughness to decrease with exposure. The region of the reactor vessel exposed to high neutron irradiation is defined as the reactor vessel beltline or

(continued)

BASES (continued)

BACKGROUND (continued)

Beltline. This is comprised of the region of the reactor vessel that directly surrounds the effective height of the active core and adjacent regions that are predicted to experience high neutron irradiation.

Linear elastic fracture mechanics (LEFM) methodology, following the guidance given by 10 CFR 50 Appendix G, ASME Section III Appendix G, and Regulatory Guide 1.99, is used to determine the stresses and material toughness at locations within the RCPB. Although any region within the pressure boundary is subject to non-ductile failure, the regions that provide the most restrictive limits are the vessel closure head flange, the feedwater nozzles, the control rod drive nozzles, and the vessel beltline.

One indicator used to indicate the temperature effect on ductility is the nil-ductility transition (NDT) temperature. The NDT temperature is a temperature below which it can be said that brittle fracture may occur. Ductile failure may occur above the NDT temperature. The NDT temperature is integrated into a reference temperature (RT_{NDT}) by testing. RT_{NDT} is a key indicator of ductility that is used for steels in pressure vessel construction. The neutron embrittlement effect on the material toughness is reflected by increasing the RT_{NDT} as exposure to neutron fluence increases. In effect the temperature at which brittle failure can occur increases.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating in accordance with ASTM E185-73 and 10 CFR 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The operating limit curves shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99 (Ref. 3 and 4).

The PT curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the geometry of the reactor vessel will dictate the most restrictive limit. Across the entire pressure/temperature span of the limit curves, some locations are more restrictive, and thus the curves are composites of the most restrictive regions.

(continued)

BASES (continued)

BACKGROUND (continued)

The curves have been developed for heatup, in-service leak and hydrostatic testing, and cooldown in conjunction with stress analyses for a large number of operating cycles and provide a conservative margin to non-ductile failure. Although they have been created to provide limits for these specific normal operations, they also can be used as a basis for determining if evaluations are necessary for abnormal transients which can begin from power operation.

APPLICABLE SAFETY ANALYSES

The limits are not derived from design basis accident analyses presented in the FSAR, but are prescribed as limits to be used during normal operation to avoid encountering pressure, temperature, and temperature rate-of-change conditions which might cause undetected flaws to propagate, in turn causing non-ductile failure of the RCPB.

RCS Pressure/Temperature Limits satisfies the requirements of the NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors (Ref. 5). While none of the three Selection Criteria (Ref. 7) directly apply, this specification preserves limits defining important boundaries for safe operation derived from the RCS stress analysis. Criterion 2 is the most appropriate criterion because operation outside of these boundaries is unanalyzed and may result in RCPB failure.

LCO

Compliance with the following PT limits is required by this LCO:

1. Operation within the PT limit curves specified in Figure B 3.4.8-1,
2. A maximum reactor coolant heatup or cooldown of 100°F in any one hour period,
3. A maximum temperature change of $\leq$ 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves in Figure B 3.4.8-1,

(continued)

BASES (continued)

LCO (continued)

4. The reactor vessel flange and head flange temperature ≥ 70°F when reactor vessel head bolting studs are under tension,

5. A temperature difference between the bottom head coolant temperature and the reactor pressure vessel coolant temperature of ≤ 100°F during recirculation pump startup, and

6. A temperature difference between the reactor coolant temperature within the recirculation loop to be started and the reactor pressure vessel coolant of ≤ 50°F during recirculation pump startup.

The above limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to non-ductile failure.

APPLICABILITY

The potential for violating the PT limits exists at all times when the reactor coolant system can be pressurized. The temperature rate of change limit can be potentially violated any time the reactor vessel is a different temperature from a cooling source.

ACTIONS

A.1, A.2

As noted, Required Actions A.1 and A.2 must be completed whenever Condition A is entered. The purpose of the Note is to give additional emphasis to the need to restore operation to the allowable condition and to also perform an evaluation of the effects of any excursion outside of the allowable limits. Restoration alone is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

Restoration within the limits is appropriate because the action is in the proper direction to reduce RCPB stress.

(continued)

BASES (continued)

ACTIONS (continued)

A.1, A.2 (continued)

The Completion Time limit of 30 minutes is based on engineering judgement. Most violations will not be so severe that the activity cannot be accomplished in this time in a controlled manner; however, if the activity cannot be accomplished, then a controlled shutdown must be initiated per Required Actions B.1 and B.2.

In addition to restoration, an evaluation to determine if RCS operation may proceed is required. The purpose of the evaluation is to determine if RCPB integrity is acceptable and must be accomplished before the event is reconciled.

If the evaluation cannot be accomplished in 72 hours, or if the results of the evaluation are indeterminate or unfavorable, then the next appropriate action is to further reduce pressure and temperature as required in Condition B.

The 72 hour Completion Time is based on engineering judgement and is reasonable to accomplish the activities necessary. For a mild violation the evaluation should be possible within this time. More severe violations may require special, event specific stress analyses and/or inspections, which are appropriately carried out while the RCS is in a reduced pressure and temperature condition as required by Condition B.

B.1, B.2

If the Required Actions and associated Completion Times are not met, a controlled shutdown must be initiated. This is a prudent action when the RCS remained in an unacceptable region for an extended period of increased stress or a sufficient severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, which is best accomplished while the RCS is in a low pressure and temperature state. With the plant at reduced pressure conditions the possibility of propagation of undetected flaws is reduced. The times allowed for a controlled shutdown to MODE 4 are reasonable and avoid placing undue stress on plant operators or plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

Verification that operation is within limits is an appropriate surveillance when RCS temperature and pressure conditions are undergoing planned changes. The time period of 30 minutes is based on engineering judgement. Since temperature rate of change limits are specified in hourly increments, a half hour time period permits assessment and correction of minor deviations within a reasonable time.

SR 3.4.8.2

A separate limit is used when the reactor is critical. Consequently, it is appropriate to verify that the RCS pressure and temperature are within the appropriate limit prior to the withdrawal of control rods that will make the reactor critical.

SR 3.4.8.3, SR 3.4.8.4

Differential temperatures within the limits of these surveillances will ensure that thermal stresses resulting from an idle recirculation pump startup will not exceed design allowances. In addition, compliance with the limit stated in SR 3.4.8.4 ensures that the assumptions of the idle recirculation loop startup analysis (Ref. 6) are satisfied. Performing the surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the surveillance and the time of the idle pump start. An acceptable means of demonstrating compliance with the 50°F requirement in SR 3.4.8.4 is to compare the temperatures between the operating recirculation loop and the idle loop.

SR 3.4.8.5

Limits on the reactor vessel flange and head flange temperature (required when the vessel head is tensioned) are sometimes bounded by the other PT limits during system heatup and cooldown. However, during operation in MODE 4, with RCS temperature less than or equal to 100°F, surveillance of the flange temperatures is required to ensure the 70°F temperature limit is not violated. With RCS temperature less than or equal to 80°F, a more frequent check of the flange temperatures is required because of the reduced margin to the limit. The flange temperatures must also be verified to be above the limit prior to and during tensioning of the vessel head bolting studs to ensure that once the head is tensioned the limit is satisfied.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Frequencies

In general, surveillance frequencies are based on industry accepted practice and engineering judgement considering the unit conditions required to perform the test, the ease of performing the test and a likelihood of a change in the system/component status.

REFERENCES

1. Code of Federal Regulations, Title 10, Part 50, Appendix G, "Fracture Toughness Requirements."

2. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure."

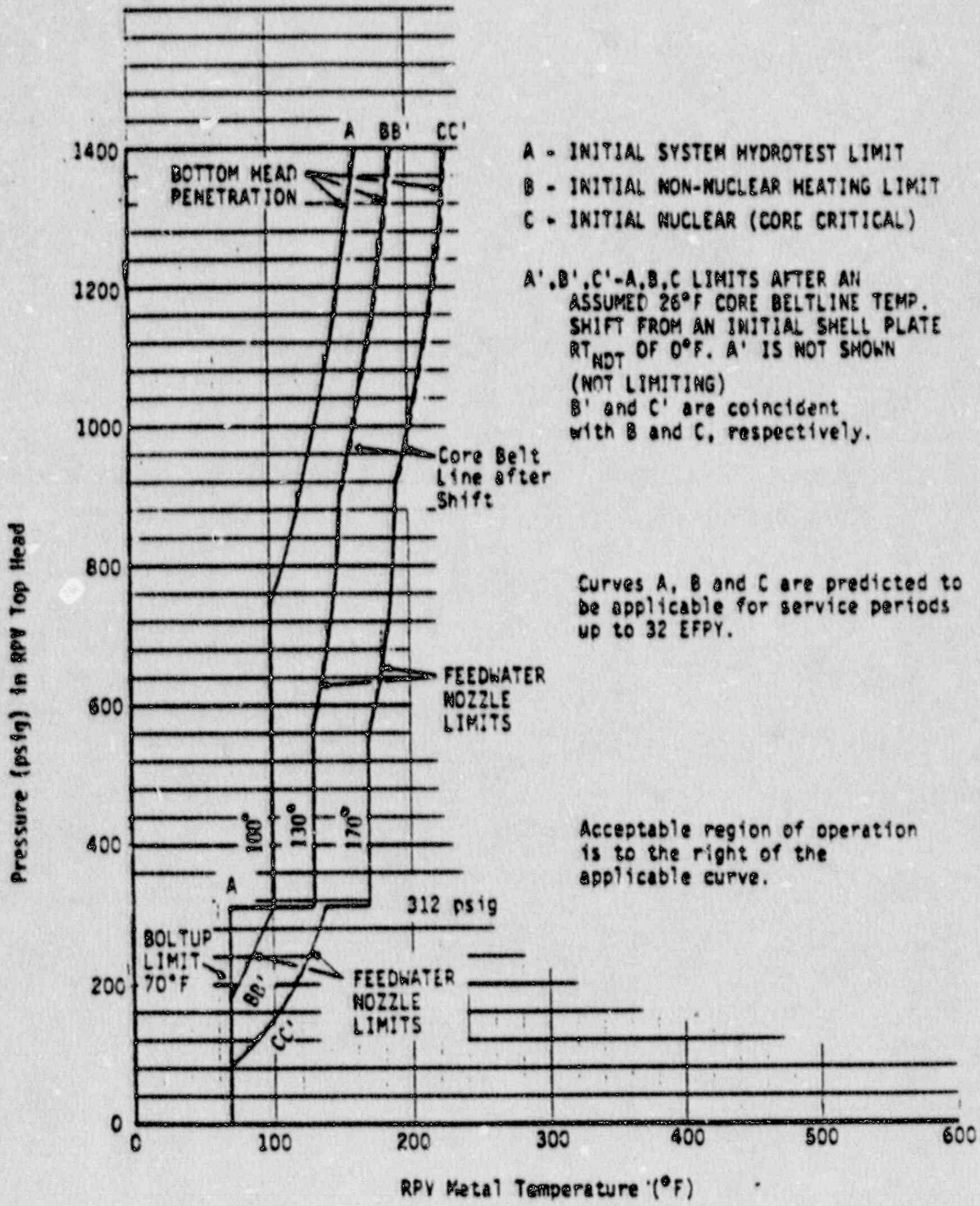
3. USNRC Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicated Radiation Damage to Reactor Vessel Materials," April 1977.

4. USNRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

5. 52FR3788, Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, USNRC, February 6, 1987.

6. Grand Gulf Unit 1 FSAR, Section 15.4.4.

7. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.



MINIMUM REACTOR PRESSURE VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE

Figure B 3.4.8-1

Grand Gulf Nuclear Station
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Revision Summary Sheet

Proposed LCO/Section: 3.4.9 Rev. 1 Steam Dome Pressure

Item	Change Description	Category
1	LCO 3.4.9 and its applicability are reformatted from LIMITING CONDITION FOR OPERATION 3.4.6.2 except as discussed below.	1
2	CONDITIONS A and B are reformatted from the ACTION statement except as discussed below.	1
3	SR 3.4.9.1 is reformatted from SR 4.4.6.2 except as discussed below.	1
4	The '*' footnote to page 3/4 4-23 is deleted because the applicability for LCO 3.4.9 includes this provision.	1
5	The CTS LCO 3.4.6.2 does not allow the pressure to be equal to 1045 psig whereas the PSTS LCO will. The REQUIRED ACTION A.1 and SR 3.4.9.1 are similarly affected.	3B

3.4 REACTOR COOLANT SYSTEM

3.4.9 Reactor Steam Dome Pressure

LCO 3.4.9 The reactor steam dome pressure shall be ≤ 1045 psig.

APPLICABILITY: MODES 1 and 2, except during anticipated transients.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure > 1045 psig.	A.1 Reduce reactor steam dome pressure to ≤ 1045 psig.	15 minutes
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify reactor steam dome pressure is ≤ 1045 psig.	12 hours

CROSS-REFERENCES: None

B 3.4 REACTOR COOLANT SYSTEM

B 3.4.9 Reactor Steam Dome Pressure

BASES

BACKGROUND	The reactor steam dome pressure is an assumed initial condition of design basis accidents and transients and is also assumed in the determination of compliance with reactor pressure vessel overpressure protection criteria.
APPLICABLE SAFETY ANALYSES	The reactor steam dome pressure is an initial condition of the vessel overpressure protection analysis of Reference 1. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system (primarily the safety/relief valves) during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure and therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved. Reference 2 also assumes an initial reactor steam dome pressure for the analysis of design basis accidents and transients used to determine the limits for fuel cladding integrity (MINIMUM CRITICAL POWER RATIO, see Bases for LCO 3.2.2 and 1% cladding plastic strain see Bases for LCO 3.2.1 and LCO 3.2.3). Reactor Steam Dome Pressure satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 3.
LCO	The specified reactor steam dome pressure limit assures the plant is operated within the assumptions of the transient analyses. Operation above the limit may result in a transient response more severe than analyzed.

(continued)

BASES (continued)

APPLICABILITY

The reactor steam dome pressure is required to be less than or equal to the limit in MODES 1 and 2 where the reactor is generating significant steam and the design basis transients and accidents are bounding. The limit may be exceeded during anticipated transients since the evaluations of References 1 and 2 demonstrate that appropriate reactor and fuel limits are not exceeded.

The limit is not applicable in MODES 3, 4 and 5, because in these modes the reactor is shutdown. The reactor pressure is well below the required limit and no anticipated events will challenge the overpressure limits.

ACTIONS

A.1, B.1

If the reactor steam dome pressure is greater than the limit, prompt action should be taken to reduce the pressure to below the limit. If the operator is unable to reduce the reactor steam dome pressure to the limit, then the reactor is required to be in MODE 3.

Completion Times

All Completion Times are based on industry accepted practice and engineering judgement considering the number of available systems and the time required to reasonably complete the Required Action.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

The reactor steam dome pressure is verified to be less than or equal to the limit every 12 hours. This ensures that the initial conditions of the design basis accidents and transients are met. Operating experience has demonstrated that the 12 hour frequency for this surveillance is adequate.

(continued)

BASES (continued)

REFERENCES

1. Grand Gulf FSAR, Section 5.2.2.

2. Grand Gulf FSAR, Section 15.

3. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

CHAPTER 3.5

ECCS AND RCIC

CHAPTER 3.5
ECCS and RCIC
TABLE OF CONTENTS

Grand Gulf Nuclear Station
Technical Specification Improvement Program

Revision Summary Sheet

Proposed LCO/Section: 3.5.1 Rev. 1 ECCS-Operating

Item	Change Description	Category
1	LCO 3.5.1 is reformatted from LIMITING CONDITION FOR OPERATION 3.5.1.	1
2	The details of system operability requirements are relocated to the BASES.	2
3	DELETED	
4	The footnote '#' to page 3/4 5-1 for applicability in MODE 2 is deleted. Special Test Exception 3.10.5 will be reviewed as a supplemental specification.	3A
5	CONDITION A is reformatted from ACTIONs a.1, a.2, and b.1.	1
6	CONDITION B was developed from ACTIONs a.3, b.2, d.1 and d.2 with the added flexibility that any two ECCS injection/spray systems may be inoperable as long as they are not LPCS and HPCS simultaneously.	3B
7	REQUIRED ACTION B.4 is added to require the remaining inoperable ECCS subsystem to be restored within 7 days.	3A+
8	CONDITION C is reformatted from ACTION c.1.	1
9	REQUIRED ACTION C.1 limits RCIC OPERABILITY to when the system is required (LCO 3.5.3).	3B
10	CONDITION D is reformatted from ACTIONs a.4, b.3, c.2 and d.3.	1
11	CONDITION E is reformatted from ACTION e.1.	1
12	CONDITION G is reformatted from ACTIONS e.1 and e.2.	1
13	SR 3.5.1.1 is reformatted from SR 4.5.1.a.1.	1
14	The method of verifying that system piping is filled in SR 3.5.1.1 is relocated.	2
15	SR 3.5.1.2 is reformatted from SR 4.5.1.a.3.	1
16	A NOTE is added to SR 3.5.1.2 to provide for LPCI OPERABILITY when in SDC mode under specified conditions.	3B+

Grand Gulf Nuclear Station
Technical Specification Improvement Program

Revision Summary Sheet

Proposed LCO/Section: 3.5.1 Rev. 1 ECCS-Operating

Item	Change Description	Category
17	SR 3.5.1 is added to define surveillance requirements for ADS.	3A+
18	SR 3.5.1.4 is reformatted from SR 4.5.1.b.	1
19	SR 3.5.1.5 is reformatted from SR 4.5.1.c.1	1
20	SR 3.5.1.6 is reformatted from SR 4.5.1.d.1.	1
21	SR 3.5.1.7 is reformatted from SR 4.5.1.d.2 and footnote '*' to page 3/4 5-5.	1
22	The method of verifying ADS valve operation in SR 3.5.1.7 is relocated.	2
23	CROSS REFERENCES are added.	1
24	Footnote '**' to page 3/4 5-1 and footnote '*' to pages 3/4 5-2 and 3/4 5-3 are deleted. The "low as practical" provision is relocated to the BASES.	2
25	LCO 3.0.2 does not perm.. more than one CONDITION for LCO 3.5.1 to be entered at a time. Therefore, the provisions in ACTIONs a, b, c, d and e to have the remaining ECCS subsystem(s) OPERABLE is not required in the CONDITIONS for LCO 3.5.1.	1
26	ACTION f is deleted.	4
27	ACTION g is deleted.	4
28	ACTION h is moved to Section 5.	1
29	ACTION i is deleted.	4
30	SR 4.5.1.a.2 is deleted. This change is considered administrative because the "keep filled" and delta P instrumentation ACTIONs are deleted (see Items 26 and 27).	4
31	SR 4.5.1.c.2 is deleted. This change is considered administrative because the "keep filled" and delta P instrumentation ACTIONs are deleted (see Items 26 and 27).	4
32	SR 4.1.5.c.3 is covered by SR 3.5.1.5 and is explicitly described in the BASES for SR 3.5.1.5.	2

Grand Gulf Nuclear Station
Technical Specification Improvement Program

Revision Summary Sheet

Proposed LCO/Section: 3.5.1 Rev. 1 ECCS-Operating

Item	Change Description	Category
33	SR 4.1.5.c.4 is deleted.	4
34	SR 4.1.5.d.3 is deleted.	4
35	SR 4.1.5.d.4 and SR 4.1.5.e are relocated. Alarm only instrumentation (except those required by RG 1.97) are not included in the Improved Tech Specs.	2
36	The '*' footnote to page 3/4 5-1 is deleted since it is now included in the applicability statement.	1
37	REQUIRED ACTION B.2 is added to require RCIC to be OPERABLE if HPCS is one of the ECCS injection/spray systems that is inoperable. (See item 6).	1+
38	REQUIRED ACTION B.1 is added to ensure that LPCS and HPCS are not simultaneously inoperable (see item 6).	1+
39	CONDITION F and REQUIRED ACTIONs F.1, F.2 and F.3 are added to allow one ADS valve and one ECCS injection/spray system to be inoperable.	3B+

3.5 ECCS AND RCIC

3.5.1 ECCS - Operating

LCO 3.5.1 All ECCS injection/spray systems shall be OPERABLE,

AND

8 ADS valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except ADS is not required to be OPERABLE with reactor steam dome pressure $\leq$ 135 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray system inoperable.	A.1 Restore inoperable system to OPERABLE status.	7 days from discovery of inoperable system
B. Two ECCS injection/spray systems inoperable.	B.1 Verify at least one ECCS spray system to be OPERABLE.	Immediately
	AND	
	B.2 ----------NOTE--------- Required Action B.2 applicable only when HPCS system is operable. ----------------------	Immediately
	Verify RCIC is OPERABLE when required to be OPERABLE.	Immediately
	AND	
	B.3 Restore at least one inoperable system to OPERABLE status.	72 hours
	AND (continued)	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	B.4	Restore the remaining inoperable system to OPERABLE status.	7 days from discovery of initial inoperable system
C. HPCS inoperable.	C.1	Verify RCIC is OPERABLE when required to be OPERABLE.	Immediately
	<u>AND</u>		
	C.2	Restore HPCS to OPERABLE status.	14 days from discovery of inoperable system
D. Required Actions and associated Completion Times of Condition A, B or C not met.	D.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	D.2	Be in MODE 4.	36 hours
E. One ADS valve inoperable.	E.1	Restore inoperable ADS valve to OPERABLE status.	14 days from discovery of inoperable valve
F. One ADS valve inoperable. <u>AND</u> One ECCS injection/spray system inoperable.	F.1	---------NOTE---------- Required Action F.1 applicable only when HPCS system is inoperable. ----------------------- Verify RCIC is OPERABLE when required to be OPERABLE.	Immediately
	<u>AND</u>	(continued)	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	F.2 Restore the inoperable injection/spray system or ADS Valve to OPERABLE status AND F.3 Restore the remaining inoperable injection/spray system or ADS valve to OPERABLE status.	72 hours 7 days from discovery of initial inoperable system/valve
G. Two or more ADS valves inoperable. OR Required Action and associated Completion Time of Condition E or F not met.	G.1 Be in MODE 3. AND G.2 Reduce reactor steam dome pressure to $\leq$ 135 psig.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.1.1	Demonstrate for each ECCS injection/spray system the system piping is filled with water from the pump discharge valve to the system isolation valve.	31 days
SR 3.5.1.2	-----------------NOTE----------------- LPCI subsystems may be considered OPERABLE during alignment to and operation in the RHR shutdown cooling mode when below the RHR cut-in permissive pressure in MODE 3, if capable of being manually realigned and not otherwise inoperable. -------------------------------------- Verify for each ECCS injection/spray system each manual, power operated or automatic valve in the flow path not locked, sealed or otherwise secured in position is in its correct position.	31 days
SR 3.5.1.3	Verify ADS air receiver pressure is ≥ 150 psig.	31 days
SR 3.5.1.4	Demonstrate the following ECCS pumps develop the specified flow rates with the specified total developed head:	According to SR 3.0.5

SYSTEM	FLOW RATE	NO. OF PUMPS	TOTAL DEVELOPED HEAD
LPCS	≥ 7115 gpm	1	≥ 290 psid
LPCI	≥ 7450 gpm	3	≥ 125 psid
HPCS	≥ 7115 gpm	1	≥ 445 psid

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.5	------------------NOTE------------------ Vessel injection may be excluded. -------------------------------------- Perform a system functional test for each ECCS injection/spray system, including simulated automatic actuation of the system throughout its emergency operating sequence, to verify each automatic valve in the flow path actuates to its correct position.	18 months
SR 3.5.1.6	-----------------NOTE----------------- Valve actuation may be excluded. -------------------------------------- Perform a system functional test for ADS, including simulated automatic actuation, throughout its emergency operating sequence.	18 months
SR 3.5.1.7	Demonstrate each ADS valve opens when manually actuated at reactor steam dome pressure $\geq$ 100 psig.	----NOTE------ Only required within 12 hours when reactor steam dome pressure is adequate to perform the test. ------------- 18 months

CROSS-REFERENCES

TITLE	NUMBER
ECCS Actuation Instrumentation	3.3.5.1
Safety/Relief Valves	3.4.4
Residual Heat Removal - Shutdown	3.4.7
Reactor Core Isolation Cooling System	3.5.3
Residual Heat Removal Suppression Pool Cooling	3.6.2.3

B 3.5 ECCS AND RCIC

B 3.5.1 ECCS - Operating

BASES

BACKGROUND

The ECCS is designed, in conjunction with the primary and secondary containment to limit the release of radioactive materials to the environment following a Loss-of-Coolant Accident (LOCA). The ECCS uses two independent methods (flooding and spraying) to cool the core during a LOCA. The ECCS injection/spray network is comprised of the High Pressure Core Spray (HPCS) system, the Low Pressure Core Spray (LPCS) system and the Low Pressure Coolant Injection (LPCI) mode of the Residual Heat Removal (RHR) system. The ECCS also consists of the Automatic Depressurization System (ADS). The suppression pool provides the source of water for the ECCS. Although no credit is taken in the safety analyses for the Condensate Storage Tank (CST) it is capable of providing a source of water for the HPCS system.

All ECCS systems are designed to ensure no single active component failure in any system will prevent automatic initiation and successful operation of the minimum required ECCS systems.

The LPCS system (Ref. 1) consists of a motor-driven pump, a spray sparger above the core, piping and valves to transfer water from the suppression pool to the sparger. The LPCS system is designed to provide cooling to the reactor core when the reactor pressure is low. Upon receipt of an initiation signal, the LPCS pump is automatically started (from normal A. C. power if available, otherwise, the pump starts after emergency A.C. power becomes available). When the reactor vessel pressure drops sufficiently, the injection valve opens and LPCS flow to the reactor vessel begins. A full-flow test line is provided to route water from and to the suppression pool to allow testing, when required, of the LPCS system without spraying water into the reactor vessel. The test return valve automatically closes on an initiation signal.

LPCI is an independent operating mode of the RHR system. There are three LPCI subsystems. Each LPCI subsystem (Ref. 2) consists of a motor-driven pump, piping and valves to transfer water from the suppression pool to the core. Each LPCI

(continued)

BASES (continued)

BACKGROUND (continued)

subsystem has its own suction and discharge piping and separate vessel nozzle which connects with the core shroud through internal piping. The LPCI subsystems are designed to provide core cooling at low reactor vessel pressure. Upon receipt of an initiation signal, each LPCI pump is automatically started (from normal A.C. power if available, otherwise, the pumps start after emergency A.C. power becomes available). When the reactor vessel pressure drops sufficiently, the injection valve opens and LPCI flow to the reactor vessel begins. With suction aligned to the suppression pool, the required RHR system valves in the LPCI flow path are automatically positioned to ensure the proper flow path for water from the suppression pool to injection into the core. The shutdown cooling and suppression pool suction valves will not automatically realign to the LPCI mode. A discharge test line is provided to route water from and to the suppression pool to allow testing, when required, of each LPCI pump without injecting water into the reactor vessel. The test return valves automatically close on an initiation signal.

The HPCS system (Ref. 3) consists of a single motor driven pump, a spray sparger above the core, and piping and valves to transfer water from the suction source to the sparger. Suction piping is provided from the CST and the suppression pool. Pump suction is normally aligned to the CST, which is the preferred source of water for injection into the RPV when HPCS functions to backup RCIC. However, if the CST water supply is low or the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a continuous water supply for continuous operation of the HPCS system. The HPCS system is designed to provide core cooling over a wide range of reactor vessel pressures 0 to 1177 psid, vessel to suction source. Upon receipt of an initiation signal, the HPCS pump automatically starts (from normal A.C. power if available, otherwise, the pump starts after emergency A.C. power becomes available) and valves in the flow path begin to align to the positions required for injection. Since the HPCS system is designed to operate over the full range of expected reactor vessel pressures, HPCS flow begins as soon as the necessary valves are open. A full-flow test line is provided to route water from and to the CST to allow testing, when required, of the HPCS system during normal operation without spraying water in the reactor vessel.

The ECCS pumps are provided with minimum flow bypass lines which discharge to the suppression pool. The valves in these lines automatically open to prevent pump damage due to overheating when other discharge line valves are closed or reactor vessel pressure is greater than the LPCS or LPCI pump discharge pressures following system initiation. To ensure

(continued)

BASES (continued)

BACKGROUND (continued)

rapid delivery of water to the reactor vessel and to minimize waterhammer effects, the ECCS discharge line keep fill systems are designed to maintain all pump discharge lines filled with water.

The ADS (Ref. 4) consists of 8 of the 20 safety/relief valves (S/RVs). It is designed to provide depressurization of the primary system during a small break LOCA if HPCS fails or is unable to maintain required water level in the reactor vessel. ADS operation reduces the reactor vessel pressure to within the operating pressure range of the low pressure ECCS systems (LPCS and LPCI), so that these systems can provide core cooling. Each ADS valve is supplied with pneumatic power from an air storage system which consists of accumulators and large air flask located in the Drywell.

APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated LOCA. The accidents for which ECCS operation is required are specifically listed in FSAR Sections 15.2.8, 15.6.4 and 15.6.5. The required analyses and assumptions are defined in Reference 5. The results of these analyses are described in Reference 6.

The ECCS system design requirements ensure the criteria of Reference 7 are satisfied under all postulated LOCA conditions assuming the worst single active component failure in the ECCS.

The limiting single failures are discussed in Reference 8. For a large break LOCA, failure of ECCS systems in Division 1 (LPCS and LPCI-A) or 2 (LPCI-B and LPCI-C) due to failure of its associated diesel generator is, in general, the most severe failure. For a small break LOCA, HPCS failure is the most severe failure. One ADS valve failure is analyzed as a limiting single failure for events requiring ADS operation. The remaining OPERABLE ECCS systems provide the capability to adequately cool the core and prevent excessive fuel damage.

ECCS-Operating satisfies the requirements of Selection Criterion 3 of NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 11.

(continued)

BASES (continued)

LCO

All ECCS injection/spray systems and 8 ADS valves are required to be OPERABLE. The ECCS injection/spray systems are defined as the three LPCI subsystems, the LPCS system and the HPCS system. The low pressure ECCS injection/spray systems are defined as the LPCS system and the three LPCI subsystems. A description of what is required for the ECCS systems to be considered OPERABLE is provided in the Background section.

With less than the required number of ECCS systems OPERABLE, the potential exists that during a limiting design basis LOCA concurrent with the worst single failure, the limits specified in Reference 7 could be exceeded. All ECCS systems must therefore be OPERABLE to satisfy the single failure criterion required by Reference 7.

A LPCI subsystem may be considered OPERABLE during alignment to and operation in the RHR shutdown cooling mode when below the RHR cut-in permissive pressure in MODE 3, if capable of being manually realigned to the LPCI mode and not otherwise inoperable. At these low pressures and decay heat levels (reactor is shutdown in MODE 3) a reduced complement of ECCS systems can provide the required core cooling thereby allowing operation of an RHR shutdown cooling loop when necessary.

APPLICABILITY

All ECCS systems are required to be OPERABLE during MODES 1, 2 and 3 when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system piping. In MODES 2 and 3 the ADS function is not required when pressure is ≤ 135 psig because the low pressure ECCS systems (LPCS, LPCI) are capable of providing flow into the reactor vessel below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2.

(continued)

BASES (continued)

ACTIONS

A.1, B.1, B.2, B.3, B.4, C.1, C.2

With one or any two (except both LPCS and HPCS) ECCS injection/spray systems inoperable, the remaining OPERABLE systems provide adequate core cooling during a LOCA. With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diverse low pressure ECCS systems in conjunction with ADS. Also, the Reactor Core Isolation Cooling (RCIC) system, for which no credit is taken in the safety analysis, will automatically provide makeup water at most reactor operating pressures. Verification of RCIC OPERABILITY is therefore required when HPCS is inoperable. This may be performed by an administrative check, by examining logs or other information, to determine if RCIC is out of service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of RCIC. However, with less than the minimum number of required ECCS injection/spray systems OPERABLE, the overall ECCS reliability is reduced because a single failure in one of the remaining subsystems concurrent with a LOCA, may result in the ECCS not being able to perform its intended safety function. Therefore, continued operation is only allowed for a limited time.

D.1, D.2

Should the Required Actions and associated Completion Times of Condition A, B or C not be met, the reactor is required to be in MODE 3 and subsequently in MODE 4. In MODE 4, the ECCS requirements are specified in LCO 3.5.2. If unable to attain MODE 4, the reactor coolant temperature should be maintained as low as practicable by use of alternate decay heat removal methods.

(continued)

BASES (continued)

ACTIONS (continued)

E.1, F.1, F.2, F.3, G.1, G.2

The LCO requires 8 ADS valves to be OPERABLE to provide the ADS function as designed. Reference 9 contains the results of an analysis which evaluated the effect of one ADS valve out of service. Per this analysis, operation of only 7 ADS valves will provide the required depressurization. However, with one ADS valve inoperable, the overall reliability of the ADS is reduced and operation is only allowed for a limited time. With one ADS valve and one ECCS system inoperable, the overall ECCS reliability is reduced because a single failure in one of the remaining systems concurrent with a LOCA may result in the ECCS not being able to perform its intended safety function. Therefore, continued operation is allowed for only a limited time. When more than one ADS valve is inoperable, system capability may not be sufficient to provided the designed function. Therefore, if the one inoperable ADS valve cannot be made OPERABLE within the allowed Completion Time or with more than one ADS valve inoperable, the reactor is required to be in MODE 3 and the reactor pressure reduced to $\leq$ 135 psig. At these conditions the ADS function is no longer required since the reactor pressure is low enough such that the low pressure ECCS subsystems can perform their designed safety function.

Completion Times

The ECCS Completion Times are based on the results of a study which evaluated the impact on ECCS unavailability assuming various components and subsystems were taken out of service. The results were used to calculate the average unavailability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowable outage times (AOT). AOTs were then chosen to provide comparable levels of ECCS availability throughout the industry (Ref. 12).

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

The pump discharge lines of the HPCS, LPCS and LPCI systems are required to be kept full with water to minimize potential waterhammer effects when the systems are initiated. Additionally, the lag between the receipt of the initiation signal and the actual injection into the reactor vessel is minimized. One acceptable method of ensuring the lines are "full" is to vent at the high points.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.2

Verification that all valves are in the required position ensures proper flow paths for ECCS. However, a valve that is capable of automatic return to its ECCS position when an ECCS signal is present, can be in position for another mode of operation. This is applicable only if the valve auto-repositions and fully opens within the time required for its ECCS function. As noted, a LPCI subsystem may be considered OPERABLE during alignment to and operation in the RHR shutdown cooling mode when below the RHR cut-in permissive pressure in MODE 3 if capable of being manually realigned to the LPCI mode and the system in not otherwise inoperable. This allows operation in the RHR shutdown cooling mode during MODE 3 if necessary.

SR 3.5.1.3

The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The designed pneumatic supply pressure requirements for the accumulator are such that following a failure of the pneumatic supply to the accumulator, at least two valve actuations can occur with the drywell at 70% of design pressure (Ref. 10). The ECCS safety analysis assumes only one actuation to achieve the depressurization required for requirement provides sufficient margin to satisfy the assumptions of the safety analyses. This minimum required pressure of 150 psig is provided by the Instrument Air System.

SR 3.5.1.4

The performance requirements of the ECCS pumps are determined through application of the 10 CFR 50 Appendix K criteria (Ref. 5). The pump flow rates, as determined by analysis, ensure that adequate core cooling is provided to satisfy the acceptance criteria of Reference 7. Periodic surveillance is performed (in accordance with ASME Section XI requirements) to verify these flow rates. The pump flow rates are verified against a system head that is equivalent to the reactor vessel pressure expected during a LOCA. The total system head developed is adequate to overcome the elevation differences between the suction source and the vessel, friction losses and pressure differences present during LOCA. These values were established during preoperational testing.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.5

The ECCS systems are required to actuate automatically to perform their design function. These surveillance tests demonstrate that with the required system initiation signals, the automatic initiation logic of HPCS, LPCS, and LPCI will cause them to operate as designed, including actuation of all automatic valves to their required position. This test also ensures the HPCS system will automatically restart on a reactor vessel low water level (Level 2) signal received subsequent to reactor vessel high water level (Level 8) trip and that the suction is automatically transferred from the CST to the suppression pool. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the reactor vessel is not required during the tests.

SR 3.5.1.6

The ADS designated S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test (logic only) is performed to demonstrate that the ADS logic operates as designed when initiated, causing proper actuation of the required components. Actual ADS valve actuation is excluded to prevent a reactor pressure vessel blowdown.

SR 3.5.1.7

A manual actuation of each ADS valve is performed to verify the valve and solenoids are functioning properly and no blockage exists in the S/RV discharge lines. This is demonstrated by the response of the turbine control or bypass valve or by a change in the measured steam flow or any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Sufficient time is therefore allowed, after the required pressure is achieved, to perform this test. Reactor startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements, prior to valve installation.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Surveillance Frequencies

In general, surveillance frequencies are based on industry accepted practice and engineering judgement considering the unit conditions required to perform the test, the ease of performing the test and a likelihood of a change in the system/component status.

REFERENCES

1. Grand Gulf FSAR, Section 6.3.2.2.3.
2. Grand Gulf FSAR, Section 6.3.2.2.4.
3. Grand Gulf FSAR, Section 6.3.2.2.1.
4. Grand Gulf FSAR, Section 6.3.2.2.2.
5. 10CFR50, Appendix K, "ECCS Evaluation Models".
6. Grand Gulf FSAR, Section 6.3.3.
7. 10CFR50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors".
8. Grand Gulf FSAR, Section 6.3.3.3.
9. Grand Gulf FSAR, Section 6.3.3.7.8.
10. Grand Gulf FSAR, Section 7.3.1.1.1.4.2.
11. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment", November 1987.
12. Memo, R. L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCO's for ECCS Components", December 1, 1975.

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Revision Summary Sheet

Proposed LCO/Section: 3.5.2 Rev. 1 ECCS-Shutdown

Item	Change Description	Category
1	LCO 3.5.2 is reformatted from LIMITING CONDITION FOR OPERATION 3.5.2.	1
2	The details of subsystem operability requirements are relocated to the BASES.	2
3	The applicability in MODE 5 is revised to eliminate the requirement to remove the transfer canal gate and to specify the water level requirement.	3B
4	CONDITION A is reformatted from ACTION a.	1
5	CONDITION B is reformatted from ACTION a except as discussed below.	1
6	CONDITION C is reformatted from ACTION b except as discussed below.	1
7	CONDITION D is reformatted from ACTION b except as discussed below.	1
8	CONDITION D requires secondary containment to be made OPERABLE as soon as practicable rather than within 8 hours per ACTION b.	3A
9	SR 3.5.2.1 is developed from LCO 3.5.3, item b, and SR 4.5.3.1.b.	3B
10	SR 3.5.2.2 is developed from SR 4.5.2.2, LCO 3.5.3 item b, and SR 4.5.3.1.	3B
11	SR 3.5.2.3 is reformatted from SR 4.5.2.1 and SR 4.5.1.a.1 except as discussed below.	1
12	SR 3.5.2.4 is reformatted from SR 4.5.2.1 and SR 4.5.1.a.3.	1
13	A NOTE to SR 3.5.2.4 is added to allow a LPCI subsystem in SDC mode to be considered OPERABLE for ECCS if capable of being manually aligned.	3B+
14	SR 3.5.2.5 is reformatted from SR 4.5.2.1 and 4.5.1.b.	1
15	SR 3.5.2.6 is reformatted from SR 4.5.2.1 and 4.5.1.c.1.	1
16	CROSS REFERENCES are added.	1

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Revision Summary Sheet

Proposed LCO/Section: 3.5.2 Rev. 1 ECCS-Shutdown

Item	Change Description	Category
17	The 3.0.4 sentence in ACTION a, along with footnote '#' to page 3/4 5-6 are deleted. Applicable only until startup from RFO3.	1
18	CONDITIONS B, C and D permit suspension of operations with a potential for draining the reactor vessel as soon as practicable versus within 4 hours and immediately of CTS ACTIONS a and b respectively.	3B
19	Suspension of CORE ALTERATIONS is deleted from ACTION b.	3B
20	The REQUIRED ACTIONS D.2, D.3 and D.4 are not equivalent to the CTS definition of SECONDARY CONTAINMENT INTEGRITY.	3B
21	The method of testing in SR 3.5.2.3 is relocated (venting at the high point vent).	3B
22	A note is added to SR 3.5.2.2 to indicate that the SR is only required when HPCS system is required to be OPERABLE.	1

3.5 ECCS AND RCIC

3.5.2 ECCS - Shutdown

LCO 3.5.2 Two ECCS injection/spray systems shall be OPERABLE.

APPLICABILITY: MODE 4,
MODE 5 except with the upper containment cavity to dryer pool gate removed and water level ≥ 22'8" over the top of the RPV flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One of the required ECCS systems inoperable.	A.1 Restore the required ECCS systems to OPERABLE status.	4 hours from discovery of inoperable system
B. Required Action and associated Completion Time of Condition A not met.	B.1 Suspend operations with a potential for draining the reactor vessel.	As soon as practicable
C. Both of the required ECCS systems inoperable.	C.1 Suspend operations with a potential for draining the reactor vessel.	As soon as practicable
	AND	
	C.2 Restore at least one ECCS system to OPERABLE status.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action C.2 and associated Completion Time not met.	D.1 Suspend operations with a potential for draining the reactor vessel. AND D.2 Ensure Secondary Containment is OPERABLE. AND D.3 Ensure the SGTS is in compliance with the requirements of Specification 3.6.4.3. AND D.4 Ensure Secondary Containment Isolation Valves are in compliance with the requirements of Specification 3.6.4.2 and Secondary Containment Actuation Instrumentation is in compliance with the requirements of Specification 3.3.6.2.	As soon as practicable

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.2.1	Verify the suppression pool water level is 12.67' when a low pressure ECCS system is required to be OPERABLE.	12 hours
SR 3.5.2.2	------------------NOTE------------------ Only required when HPCS system is required to be OPERABLE. -- Verify for the HPCS system the: A. Suppression pool water level is $\geq$ 12.67'. OR B. CST water level is $\geq$ 18'.	12 hours
SR 3.5.2.3	Demonstrate for each required ECCS injection/spray system the system piping is filled with water from the pump discharge valve to the system isolation valve.	31 days
SR 3.5.2.4	------------------NOTE------------------ LPCI subsystems may be considered OPERABLE during alignment to and operation in the RHR shutdown cooling mode if capable of being manually realigned and not otherwise inoperable. -- Verify for each required ECCS injection/spray system each manual, power operated or automatic valve in the flow path not locked, sealed or otherwise secured in position is in its correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.5	Demonstrate each required ECCS pump develops the specified flow rates with the specified total developed head. SYSTEM / FLOW RATE / NO. OF PUMPS / TOTAL DEVELOPED HEAD LPCS ≥ 7115 gpm 1 ≥ 290 psid LPCI ≥ 7450 gpm 3 ≥ 125 psid HPCS ≥ 7115 gpm 1 ≥ 445 psid	According to SR 3.0.5
SR 3.5.2.6	--------------------NOTE------------------ Vessel injection may be excluded. -- Perform a system functional test for each required ECCS injection/spray system, including simulated automatic actuation of the system throughout its emergency operating sequence, to verify each automatic valve in the flow path actuates to its correct position.	18 months

CROSS-REFERENCES

TITLE	NUMBER
ECCS Actuation Instrumentation	3.3.5.1
Secondary Containment Actuation Instrumentation	3.3.6.2
Residual Heat Removal - Shutdown	3.4.7
Secondary Containment	3.6.4.1
Secondary Containment Isolation Valves	3.6.4.2
Standby Gas Treatment System	3.6.4.3
Residual Heat Removal - High Water Level	3.9.8
Residual Heat Removal - Low Water Level	3.9.9

B 3.5 ECCS AND RCIC

B 3.5.2 ECCS - Shutdown

BASES

BACKGROUND	A description of the High Pressure Core Spray (HPCS), Low Pressure Core Spray (LPCS) system and the Low Pressure Coolant Injection (LPCI) subsystems of the Residual Heat Removal (RHR) system is provided in the Bases for LCO 3.5.1.
APPLICABLE SAFETY ANALYSIS	For MODES 1, 2 and 3 ECCS performance is evaluated for the entire spectrum of break sizes for a postulated Loss of Coolant Accident (LOCA). The long-term cooling analysis following a design basis LOCA (Ref. 1) demonstrates that only one ECCS system is required, post-LOCA, to maintain the peak cladding temperature below the allowable limit. In MODES 4 and 5, two OPERABLE ECCS systems ensure adequate vessel inventory makeup in the event of an inadvertent vessel draindown. ECCS-Shutdown satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 3.
LCO	Two ECCS injection/spray systems are required to be OPERABLE. The ECCS injection/spray systems are defined as the three LPCI subsystems, the LPCS system and the HPCS system. The LPCS system and each LPCI subsystem consists of one motor driven pump, piping and valves to transfer water from the suppression pool to the reactor vessel. The HPCS system consists of one motor driven pump, piping and valves to transfer water from the suppression pool or Condensate Storage Tank (CST) to the reactor vessel. Any LPCI subsystem (A or B) that may be aligned in the shutdown cooling mode of the RHR system in MODE 4 or 5 is considered OPERABLE for the ECCS function, if it can be manually realigned (remote or manual) to the LPCI mode and is not otherwise inoperable. Because of low pressure and temperature conditions in MODES 4 and 5, sufficient time will be available to manually align and initiate LPCI subsystem operation to provide core cooling prior to postulated fuel uncovery.

(continued)

BASES (continued)

APPLICABILITY

ECCS OPERABILITY is required in MODES 4 and 5 to assure adequate coolant inventory and sufficient heat removal capability for the irradiated fuel in the core in case of an inadvertent draindown of the vessel. Requirements for ECCS OPERABILITY during MODES 1, 2 and 3 are discussed in the Applicability section of the Bases for LCO 3.5.1. ECCS systems are not required to be OPERABLE during MODE 5 with the upper containment cavity to dryer pool gate removed and the water level maintained greater than or equal to 22'8" feet above the Reactor Pressure Vessel (RPV) flange. This provides sufficient coolant inventory to allow operator action to terminate the inventory loss prior to fuel uncovery in case of an inadvertent draindown.

The Automatic Depressurization System (ADS) is not required to be OPERABLE during MODES 4 and 5 because the reactor vessel pressure is < 135 psig and the LPCS system, HPCS system and LPCI subsystems can provide core cooling without any depressurization of the primary system being required.

ACTIONS

A.1, B.1

With one of the two required ECCS systems inoperable, the remaining OPERABLE system can provide sufficient vessel flooding capability to recover from an inadvertent vessel draindown. However, the overall system reliability is reduced because a single failure in the remaining system concurrent with a vessel draindown could result in the ECCS not being able to perform its intended function. Therefore, continued operation is only allowed for a limited time. With the inoperable system not restored to OPERABLE status within the required Completion Time, operations that have the potential for draining the reactor vessel must be suspended. This minimizes the probability of a vessel draindown and the subsequent potential for ECCS actuation.

(continued)

BASES (continued)

ACTIONS (continued)

C.1, C.2, D.1, D.2, D.3, D.4

With both of the required ECCS systems inoperable, all coolant inventory makeup capability may be unavailable and operations that have a potential for draining the reactor vessel must be suspended. If a least one ECCS system is not restored to OPERABLE status within the required Completion Times, additional actions are required to minimize any potential release of radioactive materials to the environment. This includes ensuring Secondary Containment is OPERABLE (LCO 3.6.4.1), the Standby Gas Treatment System (SGTS) is in compliance with its Specification (LCO 3.6.4.3) and the Secondary Containment Isolation Valves and Secondary Containment Actuation Instrumentation are in compliance with their Specifications (LCO 3.6.4.2 and 3.3.6.2 respectively). This may be performed by an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It does not mean to perform the surveillances needed to demonstrate OPERABILITY of the components. If however, any required component is inoperable, it must be restored to OPERABLE status. In this case, surveillance requirements may need to be performed to restore the component to OPERABLE status.

Completion Times

All Completion Times are based on industry accepted practice and engineering judgement considering the number of available systems and the time required to reasonably complete the Required Action.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1, SR 3.5.2.2

The minimum water level 12.67' required for the suppression pool is verified to ensure the suppression pool will provide adequate net positive suction head (NPSH) for the ECCS pumps, recirculation volume, vortex prevention and a safety margin for conservatism. With the suppression pool water level less than the required limit, all ECCS systems are inoperable unless they are aligned to an OPERABLE CST.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.1, SR 3.5.2.2 (continued)

When suppression pool level is less than 12.67', HPCS is considered OPERABLE only if it can take suction from the CST and the CST water level is sufficient to provide the required NPSH for the HPCS pump. Therefore, a verification that either the suppression pool water level is ≥ 12.67' or HPCS is aligned to take suction from the CST and the CST contains ≥ 170,000 gallons of water, equivalent to 18', ensures HPCS can supply makeup water to the reactor vessel.

SR 3.5.2.3, SR 3.5.2.5, SR 3.5.2.6

The bases provided for SR 3.5.1.1, SR 3.5.1.4 and SR 3.5.1.5 are applicable.

SR 3.5.2.4

Verification that all applicable valves are in the required position ensures proper flow paths for ECCS. However, a valve that is capable of automatic return to its ECCS position, when and ECCS signal is present, can be in position for another mode of operation.

In MODES 4 and 5, the RHR system may operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Therefore, during MODE 4 and 5, RHR valves that are required for LPCI subsystem operation may be aligned for the shutdown cooling mode. The LPCI mode of the RHR however, may be considered OPERABLE for the ECCS function if all the required valves in the LPCI flow path can be manually realigned to allow injection into the RPV and the system is not otherwise inoperable. This will ensure adequate core cooling if an inadvertent vessel draindown should occur.

Surveillance Frequencies

In general, surveillance frequencies are based on industry accepted practice and engineering judgement considering the unit conditions required to perform the test, the ease of performing the test and a likelihood of a change in the system/component status.

(continued)

BASES (continued)

REFERENCES

1. Grand Gulf FSAR, Section 6.3.3.4.

2. 10CFR50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors".

3. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment", November 1987.

Grand Gulf Nuclear Station
Technical Specification Improvement Program

Revision Summary Sheet

Proposed LCO/Section: 3.5.3 Rev. 1 RCIC

Item	Change Description	Category
1	LCO 3.5.3 and applicability are reformatted from LIMITING CONDITION FOR OPERATION 3.7.3 and its applicability.	1
2	The details of system operability requirements are relocated to the BASES.	2
3	DELETED	
4	CONDITIONS A and B are reformatted from the ACTION statement.	1
5	SR 3.5.3.1 is reformatted from SR 4.7.3.a.1.	1
6	The method of verifying that system piping is filled in SR 3.5.3.1 is relocated.	2
7	SR 3.5.3.2 is reformatted from SR 4.7.3.a.2.	1
8	SR 3.5.3.3 is reformatted from SR 4.7.3.b and footnote '*' to page 3/4 7-7. The pressure conditions are restated.	3B
9	SR 3.5.3.4 is reformatted from SR 4.7.3.c.2 and footnote '*' to page 3/4 7-8. The pressure conditions are restated.	3B
10	SR 3.5.3.5 is reformatted from SR 4.7.3.c.1 except as discussed below.	1
11	CROSS REFERENCES are added.	1
12	SR 4.7.3.a.3 is deleted. This is considered in the system lineup of SR 3.5.3.2 and is explicitly described in the BASES for SR 3.5.3.2.	2
13	SR 4.7.3.c.3 is deleted. This is considered in the testing of SR 3.5.3.5 and is explicitly described in the BASES for SR 3.5.3.5.	2
14	The restart requirement from SR 4.7.3.c.1 is deleted.	3B
15	The words "throughout its emergency operating sequence" are added in SR 3.5.3.5.	3A

3.5 ECCS AND RCIC

3.5.3 Reactor Core Isolation Cooling System

LCO 3.5.3 The Reactor Core Isolation Cooling (RCIC) system shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 135 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCIC inoperable.	A.1 Verify HPCS is OPERABLE. AND A.2 Restore the system to OPERABLE status.	Immediately 14 days from discovery of inoperable system
B. Required Actions and associated Completion Times of Condition A not met.	B.1 Be in MODE 3. AND B.2 Reduce reactor steam dome pressure to $\leq$ 135 psig.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Demonstrate RCIC system piping is filled with water from the pump discharge valve to the system isolation valve.	31 days
SR 3.5.3.2	Verify each manual, power operated or automatic valve in the flow path not locked, sealed or otherwise secured in position is in its correct position.	31 days
SR 3.5.3.3	Demonstrate, with reactor pressure ≤ 1045 psig, the RCIC pump can develop a flow rate ≥ 800 gpm against a system head corresponding to a reactor pressure ≥ 945 psig.	-----NOTE----- Only required within 12 hours when reactor steam dome pressure is ≥ 945 psig -------------- 92 days
SR 3.5.3.4	Demonstrate, with reactor pressure ≤ 165 psig, the RCIC pump can develop a flow rate ≥ 800 gpm against a system head corresponding to a reactor pressure ≥ 150 psig.	-----NOTE----- Only required within 12 hours when reactor steam dome pressure is ≥ 135 psig ------------ 18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.3.5	-----------------NOTE----------------- Vessel injection may be excluded. ------------------------------------- Perform a system functional test for the RCIC system, including simulated automatic actuation throughout its emergency operating sequence, to verify each automatic valve in the flow path actuates to its correct position.	18 months

CROSS-REFERENCES

TITLE	NUMBER
Reactor Core Isolation Cooling Actuation Instrumentation	3.3.5.2
ECCS - Operating	3.5.1

B 3.5 ECCS AND RCIC

B 3.5.3 Reactor Core Isolation Cooling System

BASES

BACKGROUND

NOTE: The Reactor Core Isolation Cooling (RCIC) system is not and Emergency Core Cooling System (ECCS). The RCIC system is included with the ECCS section because of similar functions during certain plant transients.

The RCIC system is designed to operate following Reactor Pressure Vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of reactor vessel water level. Under these conditions, the High Pressure Core Spray (HPCS) and RCIC systems perform similar functions.

The RCIC system (Ref. 1) consists of a steam driven turbine-pump unit, piping and valves to provide steam to the turbine, and piping and valves to transfer water from the suction source to the core via the feedwater system line. Suction piping is provided from the Condensate Storage Tank (CST) and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low or the suppression pool level high, an automatic transfer to the suppression pool assures a continuous water supply for the RCIC system. The steam supply to the turbine is piped from the main steam line A upstream of the inboard main steam line isolation valve. The RCIC system is designed to provide core cooling over a wide range of reactor pressures, 150 to 1177 psig. Upon receipt of an initiation signal, the RCIC turbine accelerates at a specified rate. As the RCIC flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow line is provided to route water from and to the CST to allow testing of the RCIC system during normal operation without injecting water into the reactor vessel.

The RCIC pump is provided with a minimum flow bypass line which discharges to the suppression pool. The valve in this line automatically opens to prevent pump damage due to overheating at reduced RCIC pump discharge flow. Low flow combined with high pump discharge pressure opens the valve.

To ensure rapid delivery of water to the reactor vessel and to minimize waterhammer effects, the RCIC system discharge line is maintained filled by the static head of water from the Condensate Storage Tank.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	The ability of the RCIC system to provide makeup coolant to the reactor is used to respond to transient events. However, the RCIC system is not an Engineered Safety Feature (ESF) system and no credit is taken in the safety analyses for RCIC system operation. However, based on its contribution to the reduction of overall plant risk, Reference 2 requires that the RCIC system be included in the technical specifications even though none of the Selection Criteria were satisfied (Ref. 3).
LCO	RCIC is required to be OPERABLE to provide makeup coolant to the reactor in the event of reactor isolation accompanied by a loss of feedwater flow. A description of what is required for the RCIC system to be considered OPERABLE is provided in the Background section.
APPLICABILITY	The RCIC system is required to be OPERABLE in MODES 1, 2 and 3 with reactor steam dome pressure > 135 psig since RCIC is the primary non-ECCS water source for core cooling when the reactor is isolated and pressurized. In MODES 2 and 3 with reactor steam dome pressure ≤ 135 psig, and in MODES 4 and 5, RCIC is not required to be OPERABLE since the ECCS subsystems can provide sufficient flow to the vessel.
ACTIONS	A.1, A.2 During MODES 1, 2 or 3 with reactor steam dome pressure > 135 psig, loss of RCIC will not affect the overall plant capability to provide makeup coolant during transients at high reactor pressure since either HPCS or RCIC is assumed to be available during plant transient analyses. OPERABILITY of HPCS is therefore verified when the RCIC system is inoperable. This may be performed by an administrative check, by examining logs or other information, to determine if the HPCS system is out of service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the HPCS system. For transients and certain abnormal events with no LOCA, RCIC (as opposed to HPCS) is the preferred source of makeup coolant because of its relatively small capacity which allows easier control of reactor vessel water level. Therefore, continued operation is only permitted for a limited time.

(continued)

BASES (continued)

ACTIONS (continued)

B.1, B.2

Should the Required Actions and associated Completion Times of Condition A not be met, the reactor is required to be in MODE 3 and the reactor pressure reduced to ≤ 135 psig. At these conditions, RCIC is no longer required.

Completion Times

The ECCS Completion times are based on the results of a study which evaluated the impact on ECCS unavailability assuming various components and subsystems were taken out of service. The results were used to calculate the average unavailability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowable outage time (AOT). AOTs were then chosen to provide comparable levels of ECCS availability throughout the industry (Ref. 4). Because of the similar functions of HPCS and RCIC, the AOTs determined for HPCS are also applied to RCIC.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The pump discharge line of the RCIC system is required to be kept full with water to minimize potential waterhammer effects when the system is initiated. Additionally, the lag between the receipt of the initiation signal and the actual injection into the reactor vessel is minimized. One acceptable method of ensuring the lines are "full" is to vent at the high points.

SR 3.5.3.2

Verification that all applicable valves are in the required position ensures proper flow paths for RCIC. For the RCIC system, this also includes the steam flow path for the turbine and the flow controller position.

SR 3.5.3.3, SR 3.5.3.4

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated and the reactor shutdown. The flow tests for the RCIC system are performed at two different pressure ranges such that system capability to provide rated flow is tested both at the higher and lower operating ranges of the system. Since the required reactor steam dome pressure must be available to perform SR 3.5.3.3 and SR 3.5.3.4, sufficient time is allowed after adequate pressure is achieved to perform these tests.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.3.5

The RCIC system is required to actuate automatically to perform its designed function. This surveillance test demonstrates with the required system initiation signals, the automatic initiation logic of RCIC will cause the system to operate as designed, including actuation of all automatic valves to their required positions. The test also ensures the RCIC system will automatically restart on a reactor vessel low water level (Level 2) signal received subsequent to a reactor vessel high water level (Level 8) trip and that suction is automatically transferred from the CST to the suppression pool. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the reactor vessel is not required during the test.

Surveillance Frequencies

In general, surveillance frequencies are based on industry accepted practice and engineering judgement considering the unit conditions required to perform the test, the ease of performing the test an a likelihood of a change in the system/component status.

REFERENCES

1. Grand Gulf FSAR, Section 5.4.6.2.
2. 52FR3788, Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.
3. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment", November 1987.
4. Memo, R. L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCO's for ECCS Components," December 1, 1975.

CHAPTER 3.7

PLANT SYSTEMS

CHAPTER 3.7
PLANT SYSTEMS
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Grand Gulf Nuclear Station
Technical Specification Improvement Program

Revision Summary Sheet

Proposed LCO/Section: 3.7.1 Rev. 1 SSW-Operating

Item	Change Description	Category
1	LCO 3.7.1 is reformatted from LIMITING CONDITIONS FOR OPERATION 3.7.1.1 and 3.7.1.3.	1
2	Details of system operability requirements are relocated to the Bases.	2
3	CONDITIONS A and B are reformatted from LCO 3.7.1.1 ACTION a except as discussed below.	1
4	REQUIRED ACTION A.1 is added to permit an individual system receiving water from SSW to be declared inoperable rather than declaring SSW in it entirety inoperable based upon the River Bend Tech Specs.	3B+
5	LCO 3.7.1.1 ACTION b for MODE 3 is deleted.	4
6	LCO 3.7.1.1 Item b.2, ACTIONS b and c, footnotes '*' and '#' to page 3/4 7-1, ACTIONS d, e, f (modes 4 and 5) and footnote '#' to page 3/4 7-2 are moved to LCO 3.7.2.	1
7	LCO 3.7.1.1 ACTION f for MODES 1,2 and 3 is deleted.	4
8	SR 3.7.1.1 is reformatted from LCO 3.7.1.3 item a and SR 4.7.1.3.a.	1
9	DELETED	
10	SR 3.7.1.2 is reformatted from SR 4.7.1.1.a.	1
11	SR 3.7.1.3 is reformatted from SR 4.7.1.3.b.	1
12	The provision to operate the fan from the control room in SR 4.7.1.3.b is deleted. The intent is to test the fan, not the control room switch.	3B
13	SR 3.7.1.4 is reformatted from SR 4.7.1.1.b and SR 4.7.1.3.c.	1
14	CROSS REFERENCES are added.	1
15	SR 3.7.1.2 has been limited to valves in lines servicing only safety related systems or components.	3B
16	Technical Specification Position Statement 101 is not incorporated into LCO 3.7.1.	4

3.7 PLANT SYSTEMS

3.7.1 Standby Service Water System - Operating

LCO 3.7.1 The Division 1 and 2 Standby Service Water (SSW) subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One of the required SSW subsystems inoperable.	A.1 Declare affected system or component inoperable. OR A.2 Restore the inoperable SSW subsystem to OPERABLE status.	Immediately upon discovery of inoperable component 72 hours from discovery of inoperable component
B. Required Actions and associated Completion Times of Condition A not met. OR Both SSW subsystems inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.1.1	Verify the ultimate heat sink basin water level is ≥ 7.25'. ~~feet~~.	24 hours
SR 3.7.1.2	Verify for each required SSW subsystem each manual, power operated or automatic valve in SSW flow paths servicing safety related systems or components not locked, sealed or otherwise secured in position is in its correct position.	31 days
SR 3.7.1.3	Demonstate each cooling tower fan operates for ≥ 15 minutes.	31 days
SR 3.7.1.4	Perform a system functional test for each required SSW subsystem including simulated automatic actuation of the subsystem.	18 months

CROSS-REFERENCES

TITLE	NUMBER
Residual Heat Removal - Shutdown	3.4.7
ECCS - Operating	3.5.1
Reactor Core Isolation Cooling System	3.5.3
Residual Heat Removal Containment Spray	3.6.1.9
Residual Heat Removal Suppression Pool Cooling	3.6.2.3
A.C. Sources - Operating	3.8.1

B 3.7 PLANT SYSTEMS

B 3.7.1 Standby Service Water System - Operating

BASES

BACKGROUND

The Standby Service Water (SSW) System is designed to provide cooling water for the removal of heat from plant auxiliaries, such as Residual Heat Removal (RHR) system heat exchangers, standby diesel generators, and room coolers for Emergency Core Cooling System (ECCS) equipment, required for a safe reactor shutdown following a design basis transient or accident. The SSW system also provides cooling to plant components, as required, during normal shutdown and reactor isolation modes. During a design basis accident, the equipment required for normal operation only, is isolated from the SSW system and cooling is directed only to safety related equipment.

For the purpose of this technical specification the SSW system consists of the ultimate heat sink (UHS), two independent cooling water headers (subsystems A and B) and their associated pumps, piping, valves and instrumentation. Subsystems A and B are redundant and service equipment in Division I and II, respectively.

The UHS is two concrete makeup water basins each comprised of one cooling tower with four independent fan cells (two fan cells per basin). The combined basin volume is sized such that sufficient water inventory is available for all SSW system post-LOCA cooling requirements for a 30 day period with no external makeup water source available (Ref. 1). Normal makeup for each basin is provided automatically by the Plant Service Water (PSW) System.

Cooling water is pumped from the cooling tower basins by the two SSW pumps to the essential components through the two main redundant supply headers (Subsystems A and B). After removing heat from the components, the water is discharged to the cooling towers where the heat is rejected through direct contact with ambient air.

(continued)

BASES (continued)

BACKGROUND (continued)

Subsystems A and B supply cooling water to redundant equipment required for a safe reactor shutdown. The specific equipment for which the SSW system supplies cooling water is listed in Reference 2. Subsystem A and B pumps are sized such that the operation of both of them or one of them in conjunction with the HPCS Service Water System (LCO 3.7.3) pump will provide adequate cooling water to the equipment required for safe shutdown. The SSW system is designed to withstand a single active or passive failure coincident with a loss of offsite power without losing the capability to supply adequate cooling water to equipment required for safe reactor shutdown (Ref. 3). Following a design basis accident or transient, the SSW system will operate automatically and without operator action. Manual initiation of supported systems, e.g. suppression pool cooling, is however, performed for long term cooling operations.

APPLICABLE SAFETY ANALYSES

The ability of the SSW system to support long term cooling of the reactor or containment is evaluated in FSAR Chapters 6 (Engineered Safety Features), 9 (Auxiliary Systems) and 15 (Accident Analyses). These analyses explicitly assume the SSW will provide adequate cooling support to the equipment required for safe reactor shutdown. These analyses include the evaluation of the long term containment response after a design basis Loss Of Coolant Accident (LOCA). The SSW system provides cooling water for the RHR suppression pool cooling mode to limit the suppression pool temperature and containment pressure following a LOCA. This ensures the containment can perform its intended function of limiting the release of radioactive materials to the environment following a LOCA. The SSW system also provides cooling to other components assumed to function during a LOCA.

The safety analyses for long term cooling were performed (Ref. 4 and 5) for a LOCA concurrent with a loss of offsite power, and minimum available diesel generator power. The worst case single failure which would affect the performance of the SSW subsystems A or B is the failure of one of the two standby diesel generators which would affect one subsystem of the SSW system. The SSW flow assumed in the analyses is 7900 gpm per pump (Ref. 6).

SSW System - Operating satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 6.

(continued)

BASES (continued)

LCO

The OPERABILITY of Division I (subsystem A) and Division II (subsystem B) of the SSW system is required to ensure the effective operation of the RHR system in removing heat from the reactor and the effective operation of other safety related equipment during a design basis accident or transient. The OPERABILITY of each independent subsystem of the SSW system is based on having an OPERABLE UHS, the pump in the subsystem OPERABLE and an OPERABLE flow path capable of taking suction from the associated SSW cooling basin and transferring the water to the appropriate plant equipment, as required. Requiring both subsystems OPERABLE assures either subsystem A or B subsystems A and B together will be available to provide adequate capability to meet cooling requirements of the equipment required for safe shutdown.

The OPERABILITY of the UHS is based on having a minimum basin water level at or above elevation 130' 3" mean sea level (which is equivalent to an indicated level of $\geq$ 7.25 feet), and having four OPERABLE cooling tower fans.

APPLICABILITY

The requirements for OPERABILITY of the SSW system including the cooling tower basins in MODES 1, 2, and 3 are governed by the required OPERABILITY of the equipment serviced by the SSW system in those MODES. SSW system requirements for other operating modes are covered in LCO 3.7.2.

ACTIONS

A.1, A.2

With SSW subsystem A or B inoperable either due to an inoperable pump or inoperable flow path, sufficient cooling water can be supplied by the remaining OPERABLE subsystem should a reactor shutdown be necessary. However, if an additional single failure in the SSW system were to occur, the system would not be capable of performing its intended function. Therefore, only a limited time is allowed to restore the inoperable subsystem to OPERABLE status. Alternatively, when the SSW flow path to any safety related system or component is inoperable, the affected system may be declared inoperable and the applicable LCO is entered to determine the appropriate action. The SSW subsystem may still be capable of providing cooling water to all other associated systems.

(continued)

BASES (continued)

ACTIONS (continued)

B.1, B.2

If the Required Actions and associated Completion Times of Condition A cannot be met, the reactor is required to be in MODE 3 in 12 hours and in MODE 4 in the following 24 hours. In MODE 4 the system requirements are reduced as specified in LCO 3.7.2.

Additionally, with both subsystems of the SSW system inoperable, the associated equipment cannot perform the intended function. Continued operation in these MODES cannot be justified. Therefore, the reactor is required to be in MODE 3 and subsequently in MODE 4. If MODE 4 cannot be achieved because of the inoperable SSW subsystems, the reactor coolant temperature should be maintained as low as practicable using an alternate decay heat removal method.

Completion Times

All Completion Times are based on industry accepted practice and engineering judgement considering the number of available systems and the time required to reasonably complete the Required Action.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This surveillance verifies the cooling tower basins have sufficient cooling water (as measured by basin water level) to satisfy the design basis of 30 day cooling capability with no external makeup source. With the ultimate heat sink inoperable, the affected SSW subsystems must be declared inoperable.

SR 3.7.1.2

Verification of the correct alignment of all applicable valves is essential to ensure the proper flow paths servicing safety related systems or components for the SSW subsystems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.1.3

This surveillance verifies the OPERABILITY of the SSW cooling tower fans. The 15 minute duration for fan operation is sufficient to monitor the steady state performance of the fans.

SR 3.7.1.4

This surveillance verifies the automatic isolation valves of the SSW system will automatically switch to the safety or emergency position to provide cooling water exclusively to the safety related equipment during an accident event. This surveillance also verifies the automatic start capability of the SSW cooling tower fans. The fans are required to start automatically whenever the associated SSW subsystem is started.

Surveillance Frequencies

In general, Surveillance Frequencies are based on industry accepted practice and engineering judgement considering the unit conditions required to perform the test, the ease of performing the test and a likelihood of a change in the system/component status.

REFERENCES

1. Grand Gulf FSAR, Section 9.2.5.1.1.
2. Grand Gulf FSAR, Table 9.2-3.
3. Grand Gulf FSAR, Section 9.2.1.1.1.a and 9.2.1.1.1.d.
4. Grand Gulf FSAR, Section 6.2.1.1.3.3.1.6.
5. Grand Gulf FSAR, Section 6.2 2.3.
6. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment", November 1987.

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Revision Summary Sheet

Proposed LCO/Section: 3.7.2 Rev. 1 SSW-Shutdown

Item	Change Description	Category
1	LCO 3.7.2 is reformatted from LIMITING CONDITIONS FOR OPERATION 3.7.1.1 and 3.7.1.3.	1
2	Details of system operability requirements are relocated to the Bases.	2
3	CONDITION A is reformatted from LCO 3.7.1.1 ACTIONS b, c, d, e and f and LCO 3.7.1.3 ACTION c.	1
4	SR 3.7.2.1 is reformatted from SR 4.7.1.3.a and LCO 3.7.1.3 item a.	1
5	DELETED	
6	SR 3.7.2.2 is reformatted from SR 4.7.1.1.a except as discussed below.	1
7	SR 3.7.2.3 is reformatted from SR 4.7.1.3.b except as discussed below.	1
8	The provision to start the fan from the control room in SR 4.7.1.3.b is deleted. Intent is to test the fan not the control room handswitch.	3B
9	SR 3.7.2.4 is reformatted from SR 4.7.1.1.b and SR 4.7.1.3.c.	1
10	CROSS REFERENCES are added.	1
11	The Specification 3.0.4 exceptions in LCO 3.7.1.1 ACTIONS b, c, and d, the Specification 3.0.3 exception in ACTION e and the '#' footnotes to pages 3/4 7-1 and 3/4 7-2 are deleted as they are no longer applicable.	1
12	Footnote '**' to page 3/4 7-1 is deleted. The alternate MODE 4 provision is relocated to the Bases.	2
13	Footnote '*' to page 3/4 7-1 and 3/4 7-4 is relocated to PSTS 3.6.4.1.	2
14	The Specification 3.0.4 exception in LCO 3.7.1.3 ACTION a, footnote '**' and Specification 3.0.3 exception in Action c are deleted as they are no longer applicable.	1
15	Footnote '##' to page 3/4 7-4 is relocated to the Bases.	2

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Proposed LCO/Section: 3.7.2 Rev. 1 SSW-Shutdown

Item	Change Description	Category
16	SR 3.7.2.2 has been limited to valves in lines servicing only safety related systems or components.	3B
17	Footnote '#' to page 3/4 7-1 is relocated to Bases 3.7.3.	2

3.7 PLANT SYSTEMS

3.7.2 Standby Service Water System - Shutdown

LCO 3.7.2 The Division 1 or 2 Standby Service Water (SSW) subsystem shall be OPERABLE.

APPLICABILITY: MODES 4 and 5,
When associated systems and components are required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required SSW subsystem inoperable.	A.1 Declare affected system or component inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Verify the ultimate heat sink basin water level is $\geq$ 7.25' ~~feet~~.	24 hours
SR 3.7.2.2	Verify for each required SSW subsystem each manual, power operated or automatic valve in each required SSW flow path servicing safety related systems or components not locked, sealed or otherwise secured in position is in its correct position.	31 days
SR 3.7.2.3	Demonstrate each cooling tower fan operates for $\geq$ 15 minutes.	31 days
SR 3.7.2.4	Perform a system functional test for each required SSW subsystem including simulated automatic actuation of the subsystem.	18 months

CROSS-REFERENCES

TITLE	NUMBER
Residual Heat Removal - Shutdown	3.4.7
ECCS - Shutdown	3.5.2
A.C. Sources - Shutdown	3.8.2
Residual Heat Removal - High Water Level	3.9.8
Residual Heat Removal - Low Water Level	3.9.9

B 3.7 PLANT SYSTEMS

B 3.7.2 Standby Service Water System - Shutdown

BASES

BACKGROUND

The Standby Service Water (SSW) System is described in the Bases for LCO 3.7.1.

APPLICABLE SAFETY ANALYSES

The ability of the SSW system to support long term cooling of the reactor or containment is evaluated in FSAR Chapters 6 (Engineered Safety Features), 9 (Auxiliary Systems) and 15 (Accident Analyses). These analyses explicitly assume that the SSW will provide adequate cooling support to the equipment required for safe reactor shutdown. These analyses include the evaluation of the long term containment response after a design basis accident (DBA).

During shutdown, refueling, or fuel handling conditions equipment cooling support may be required. Safety analyses assume that the SSW system will support components and systems required to maintain core cooling following a loss of offsite power or loss of normal shutdown cooling capability occurring while in MODES 4 or 5. During fuel handling or handling of loads which, if dropped, could result in release of radioactive material, SSW must also be available to provide cooling to the Standby Diesel Generations in the event of a loss of offsite power. This is required to maintain the Standby Gas Treatment system in operation for secondary containment integrity.

Should the reactor vessel inadvertently draindown, cooling support will also be required for ECCS operation.

SSW System - Shutdown satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 1.

(continued)

BASES (continued)

LCO

The OPERABILITY of the SSW system is required to ensure the effective operation of the Residual Heat Removal (RHR) system in removing heat from the reactor and the effective operation of other safety related equipment during a design basis accident or transient. The OPERABILITY of each independent subsystem of the SSW system is based on having an OPERABLE UHS, an OPERABLE pump in the subsystem, and an OPERABLE flow path capable of taking suction from the associated SSW cooling basin and transferring the water to the appropriate plant equipment, as required. Only those subsystems associated with the systems and components required to be OPERABLE by LCO 3.4.7, LCO 3.5.2, LCO 3.8.2, LCO 3.9.8 and LCO 3.9.9 are required to be OPERABLE in MODES 4 and 5.

The OPERABILITY of the UHS is based on having a minimum basin water level at or above elevation 130′ 3" mean sea level (which is equivalent to an indicated level of ≥ 7.25 feet) and having two OPERABLE cooling tower fans for each subsystem required to be OPERABLE.

APPLICABILITY

The requirements for OPERABILITY of the SSW system in MODES 4 and 5 are governed by the required OPERABILITY of the equipment serviced by the SSW system in those MODES. SSW system requirements for MODES 1,2, and 3 are specified in LCO 3.7.1.

ACTIONS

A.1

With the required SSW subsystem inoperable, the capability of the affected systems to perform their intended functions cannot be assured. Therefore, the affected systems are required to be declared inoperable and the Required Actions specified in the appropriate LCOs followed.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1, SR 3.7.2.2, SR 3.7.2.3, SR 3.7.2.4

The Bases provided for SR 3.7.1.1 through SR 3.7.1.4 are applicable.

Surveillance Frequencies

In general, Surveillance Frequencies are based on industry accepted practice and engineering judgement considering the unit conditions required to perform the test, the ease of performing the test and a likelihood of a change in the system/component status.

REFERENCES

1. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

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Revision Summary Sheet

Proposed LCO/Section: 3.7.3 Rev. 1 HPCS Service Water

Item	Change Description	Category
1	LCO 3.7.3 is reformatted from LIMITING CONDITIONS FOR OPERATION 3.7.1.2 and 3.7.1.3.	1
2	Details of system operability requirements are relocated to the Bases.	2
3	The applicability is revised to be when HPCS is required to be OPERABLE.	3B
4	CONDITION A is reformatted from the LCO 3.7.1.2 ACTION statement and LCO 3.7.1.3 ACTIONS a and b.	1
5	CONDITION A does not require the HPCS associated diesel generator to be declared inoperable.	3B
6	SR 3.7.3.1 is reformatted from SR 4.7.1.3.a.	1
7	DELETED	
8	SR 3.7.3.2 is reformatted from SR 4.7.1.2.a.	1
9	SR 3.7.3.3 is reformatted from SR 4.7.1.2.b and SR 4.7.1.3.c.	1
10	CROSS REFERENCES are added.	1
11	The references in the ACTION statement to 3.5.1 and 3.5.2 are deleted.	1

3.7 PLANT SYSTEMS

3.7.3 HPCS Service Water System

LCO 3.7.3 The HPCS Service Water System shall be OPERABLE.

APPLICABILITY: When the HPCS system is required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. HPCS Service Water System inoperable.	A.1 Declares HPCS inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.3.1	Verify the ultimate heat sink basin water level is $\geq$ 7.25'.	24 hours
SR 3.7.3.2	Verify each manual, power operated or automatic valve in the HPCS service water flow path servicing safety related equipment not locked, sealed or otherwise secured in position is in its correct position.	31 days
SR 3.7.3.3	Perform a system functional test including simulated automatic actuation of the system.	18 months

CROSS REFERENCES

TITLE	NUMBER
ECCS - Operating	3.5.1
ECCS - Shutdown	3.5.2
SSW - Operating	3.7.1
SSW - Shutdown	3.7.2

B 3.7 PLANT SYSTEMS

B 3.7.3 <u>HPCS Service Water System</u>

BASES

BACKGROUND

The High Pressure Core Spray (HPCS) Service Water System is designed to provide cooling water for the removal of heat from essential support components of the HPCS system.

For the purpose of this technical specification the HPCS Service Water system consists of the ultimate heat sink (UHS), one cooling water header (subsystem C of the SSW system), the HPCS Service Water Pump and the associated piping and valves.

The UHS for the HPCS Service Water System is described in the Bases for LCO 3.7.1.

Cooling water is pumped from cooling tower Basin "A" by the HPCS service water pump to the essential Division 3 support components through the HPCS service water supply header (Subsystem C). After removing heat from the components, the water is discharged to the cooling towers where the heat is rejected through direct contact with ambient air.

The HPCS service water system specifically supplies cooling water to the HPCS diesel generator jacket water coolers and HPCS pump room cooler. The HPCS service water system pump is sized such that it will provide adequate cooling water to the Division 3 equipment required for safe shutdown. Following a design basis accident or transient, the HPCS service water system will operate automatically.

APPLICABLE SAFETY ANALYSIS

The ability of the HPCS service water system to support both short term and long term cooling of the reactor following a DBA is evaluated in FSAR Chapters 6 (Engineered Safety Features), 9 (Auxiliary Systems) and 15 (Accident Analyses). These analyses explicitly assume the HPCS Service Water System will provide adequate cooling support to Division 3 equipment required for safe reactor shutdown.

HPCS service water system satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 2.

(continued)

BASES (continued)

LCO

The OPERABILITY of the HPCS service water system is based on having an OPERABLE UHS, an OPERABLE pump and an OPERABLE flow path capable of taking suction from the associated SSW cooling basin and transferring the water to the appropriate plant equipment, as required. Requiring the HPCS service water system OPERABLE assures that HPCS service water system will be available to provide adequate capability to meet cooling requirements of the Division 3 ESF equipment required for safe shutdown.

The OPERABILITY of the UHS is based on having a minimum basin water level at or above evaluation 130'3" mean sea level (which is equivalent to an indicated level of ≥ 7.25 feet).

APPLICABILITY

The requirements for OPERABILITY of the HPCS service water system including the cooling tower basins are governed by the required OPERABILITY of the Division 3 ESF equipment serviced by the HPCS service water system.

ACTIONS

A.1

When the HPCS service water system is inoperable, the capability of the HPCS system to perform its intended function cannot be assured. Therefore, the HPCS system is required to be declared inoperable and the Required Actions specified in the appropriate LCO followed.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This surveillance verifies that the associated cooling tower basin has sufficient cooling water inventory (as measured by basin water level) to satisfy the design basis cooling capability for Division 3 components during a Design Basis Accident. SR 3.7.1.1 addresses this requirement for SSW Division 1 and 2. With the ultimate heat sink inoperable, the HPCS system must be declared inoperable.

SR 3.7.3.2

Verification of the correct alignment of all applicable valves is essential to ensure the proper flow paths servicing safety related systems or components for the HPCS system.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.3.3

This surveillance verifies that the automatic valve in the HPCS service water system will automatically move to its emergency position and that the HPCS service water pump will automatically start to provide flow to its safety related equipment during an accident.

Surveillance Frequencies

In general, Surveillance Frequencies are based on industry accepted practice and engineering judgement considering the unit conditions required to perform the test, the ease of performing the test and a likelihood of a change in the system/component status.

REFERENCES

1. Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants", Para. C.1.

2. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

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Revision Summary Sheet

Proposed LCO/Section: 3.7.4 Rev. 1 Control Room Fresh Air System

Item	Change Description	Category
1	LCO 3.7.4 is reformatted from LIMITING CONDITION FOR OPERATION 3.7.2.	1
2	Footnote '*' to page 3/4 7-5 is deleted. The applicability statement in LCO 3.7.4 directly includes this requirement.	1
3	A NOTE is added to indicate CONDITIONS may be concurrently applicable.	1
4	CONDITIONS A and B are reformatted from ACTION a.	1
5	CONDITIONS A and C are reformatted from ACTIONS b.1 and b.2 except as discussed below.	1
6	The Specification 3.0.4 exception in ACTION b.1 and footnote '#' to page 3/4 7-5 are deleted as being no longer applicable.	1
7	CONDITION B adds actions to be taken when both subsystems are inoperable in MODES 1, 2 and 3.	3B+
8	CONDITION C permits REQUIRED ACTIONS C.2.1, C.2.2 and C.2.3 to be done instead of REQUIRED ACTION C.1. ACTION b.1 did not provide this portion.	3B
9	The NOTE added to REQUIRED ACTION C.2.2 is reformatted from ACTION c.	1
10	COMPLETION TIMES are provided for CONDITION c. Times were previously unstated.	3A
11	SR 3.7.4.1 is reformatted from SR 3.7.2.a except as discussed below.	1
12	The STAGGERED TEST BASIS of SR 4.7.2.a is deleted. The Comparison Document discussion for LCO 3.7.4 characterizes such testing as having negative benefits in justifying deleting the requirement. NOTE: The Improved Tech Specs adds the STAGGERED TEST BASIS for some other LCOs.	3B
13	SR 3.7.4.2 is developed from SR 4.7.2.b.2 and SR 4.7.2.e.	3B
14	SR 3.7.4.3 is developed from SR 4.7.2.b.2 and SR 4.7.2.f.	3B

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Revision Summary Sheet

Proposed LCO/Section: 3.7.4 Rev. 1 Control Room Fresh Air System

tem	Change Description	Category
15	SR 3.7.4.4 is reformatted from SR 4.7.2.b.3 and SR 4.7.2.c except as discussed below.	1
16	The frequency for SR 3.7.4.4 is reduced to 1440 hours of charcoal adsorber operation from 720 hours based upon industry experience.	3B
17	SR 3.7.4.5 is reformatted from SR 4.7.2.d.1.	1
18	SR 3.7.4.6 is developed from SR 4.7.2.d.3.	3B
19	SR 3.7.4.7 is developed from SR 4.7.2.d.2.	3B
20	Actuation instrumentation in SR 4.7.2.d.2 is moved to LCO 3.3.7.1.	1
21	SR 4.7.2.b.4 is verified during performance of SR 3.7.4.2, SR 3.7.4.3 and SR 3.7.4.5.	3B
22	Details for system operability are removed from LCO 3.7.2 and relocated to the Bases.	2
23	SR 3.7.4.1 deletes requirements to initiate from the control room the CRFA subsystems currently in SR 4.7.2.a.	4
24	SR 3.7.4.1 deletes requirements to have subsystem run at least 10 hours continuously currently in CTS SR 4.7.2.a.	4
25	Requirements to perform SR 4.7.2.b after any structural maintenance in the HEPA filter or charcoal adsorber housing is deleted.	4
26	The methods and acceptance criteria of SR 4.7.2.b.3 and 4.7.2.c are relocated.	2
27	The control room manual initiation test condition is deleted from SR 4.7.2.d.2.	4

3.7 PLANT SYSTEMS

3.7.4 Control Room Fresh Air System

LCO 3.7.4 Two Control Room Fresh Air (CRFA) subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, and 5,
When handling irradiated fuel ~~or suspended light loads over irradiated fuel~~ in the primary or secondary containment.

------------------------NOTE------------------------
Conditions A, B and C may be concurrently applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRFA subsystem inoperable.	A.1 Restore the inoperable subsystem to OPERABLE status.	7 days from discovery of inoperable subsystem
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3. OR Both CRFA subsystems inoperable in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 4, 5, or when handling irradiated fuel ~~or suspended light loads over irradiated fuel~~ in the primary or secondary containment. OR Both CRFA subsystems inoperable in MODE 4, 5, or when handling irradiated fuel ~~or suspended light loads over irradiated fuel~~ in the primary or secondary containment.	C.1 Place OPERABLE subsystem in the isolation mode of operation.	Immediately
	OR	
	C.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.2.2 ---------NOTE---------- Provisions of LCO 3.0.3 are not applicable. ---------------------- Suspend handling of irradiated fuel in the primary and secondary containment.	Immediately
	AND	
	C.2.3 Suspend operations with a potential for draining the reactor vessel.	As soon as practicable
	~~AND~~	
	~~C.2.4 Suspend handling light loads over irradiated fuel.~~	~~As soon as practicable~~

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.4.1	Demonstrate each subsystem operates with flow through the HEPA filters and charcoal adsorbers for ≥ 10 hours with the heaters operable.	31 days
SR 3.7.4.2	Demonstrate < 0.05% penetration of the HEPA filters by a DOP test at a system flow rate of 3600 to 4400 cfm.	18 months AND Once within 7 days after painting, fire, or chemical release in the area being serviced by the filter AND Prior to declaring subsystem OPERABLE after each complete or partial replacement of a filter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.4.3	Demonstrate < 0.05% bypass leakage through the adsorber section by a halogenated hydrocarbon test at a system flow rate of 3600 to 4400 cfm.	18 months AND Once within 7 days after painting, fire, or chemical release in the area being serviced by the filter AND Prior to declaring subsystem OPERABLE after each complete or partial replacement of an adsorber bank

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.7.4.4	-----------------NOTE----------------- Analysis must be completed within 31 days of sampling. -------------------------------------- Remove and perform a laboratory analysis of a representative charcoal adsorber sample for methyl iodide, penetration.	1440 hours of charcoal adsorber operation AND 18 months AND Once within 7 days after painting, fire, or chemical release in the area being serviced by the filter
SR 3.7.4.5	Demonstrate < 7.2 inches water gauge pressure drop across the combined HEPA filters and charcoal adsorber banks at a system flow rate of 3600 to 4400 cfm.	18 months
SR 3.7.4.6	Demonstrate heaters dissipate from 18.6 to 22.8 kw.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.4.7	Demonstrate each subsystem automatically switches to the isolation mode of operation on receipt of an actuation signal.	18 months

CROSS-REFERENCES: None

B 3.7 PLANT SYSTEMS

B 3.7.4 **Control Room Fresh Air System**

BASES

BACKGROUND

The Control Room Fresh Air (CRFA) system is designed to provide a radiologically controlled environment to ensure the control room will remain habitable for personnel during and following a design basis accident (DBA) (Ref. 1). To meet these requirements, this system is designed in conjunction with control room design provisions such that the radiation exposure to personnel inside the control room is less than 5 rem whole body consistent with the requirements of General Design Criterion 19 of Appendix A to 10 CFR 50.

The safety related function of the CRFA system used to control radiation exposure consists of two independent and redundant high efficiency air filtration subsystems. Each subsystem consists of a demister, an electric heater, prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter and a fan.

In addition to the safety related standby emergency filtration function, parts of the CRFA system are operated to maintain the control room environment during normal operation. Upon receipt of an actuation signal (indicative of conditions that could result in radiation exposure to control room personnel) the CRFA system automatically switches to the isolation mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room and control room air flow is recirculated and processed through either of the two filter subsystems. When conditions permit, fresh air can be manually brought into the control room through the charcoal filter system.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ability of the CRFA system to maintain the habitability of the control room is an explicit assumption for the safety analyses evaluated in FSAR Chapters 6, (Engineered Safety Features) and 15, (Accident Analyses). The isolation mode of the CRFA system is assumed to operate following a Loss Of Coolant Accident (LOCA), main steam line break (MSLB), fuel handling accident and control rod drop accident (CRDA). The radiological doses to control room personnel as a result of the various design basis accidents are summarized in Reference 2. In all cases, the doses are within the limits of 10 CFR 50, Appendix A, General Design Criterion 19.

Control Room Fresh Air System satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 5.

LCO

Two redundant subsystems of the CRFA system are required to ensure at least one is available assuming a single failure disables the other subsystem. The CRFA subsystem consists of two independent and redundant filtration trains. Should any component in one subsystem fail, filtration can be performed by the other subsystem. The OPERABILITY of each independent subsystem is based on having adequate system flow and OPERABLE HEPA filters, charcoal adsorbers and heaters. A description of what is required for CRFA to be considered OPERABLE is provided in the Background Section.

APPLICABILITY

The standby emergency filtration portion of the CRFA system is required to be OPERABLE in MODES 1, 2, 3, 4, 5 and when handling irradiated fuel ~~or when handling suspended light loads over irradiated fuel~~ in the primary or secondary containment to ensure the control room will be habitable for personnel during and following a design basis accident.

(continued)

BASES (continued)

ACTIONS

A.1

With one CRFA subsystem inoperable, the remaining OPERABLE subsystem can maintain the habitability of the control room during the postulated design basis accidents assuming no additional failures in the OPERABLE subsystem. However, if a single active component fails concurrent with the postulated design basis accident, depending on the specific failure, the CRFA system may not be able to perform its intended safety function. Therefore, system reliability is reduced and operation is only allowed to continue for a limited time.

B.1, B.2

In MODES 1, 2 and 3, with an inoperable subsystem not restored to OPERABLE status and the associated Completion Time not met or both subsystems inoperable, the CRFA system may not be capable of performing its intended safety function and the reactor is required to be in MODE 3 and subsequently in MODE 4.

C.1, C.2.1, C.2.2, C.2.3

In MODES 4, 5 and when handling irradiated fuel ~~or suspended light loads over irradiated fuel~~ in the primary or secondary containment, if an inoperable subsystem cannot be restored to OPERABLE status and the associated Completion Time is not met, the remaining OPERABLE subsystem may be placed in the isolation mode of operation. This ensures the CRFA system is operating prior to any potential design basis accident that could require the CRFA system to actuate. This action provides a continuous check of the operation of the CRFA system. Alternatively, CORE ALTERATIONS, handling of irradiated fuel ~~or suspended light loads over irradiated fuel~~ in the primary or secondary containment and operations that could drain the reactor vessel must be suspended. This eliminates the potential accidents under these conditions that could require the CRFA system to function. Suspension of these activities shall not preclude completion of the movement of a component to a safe, conservative position.

Completion Times

All Completion Times are based on industry accepted practice and engineering judgement considering the number of available systems and the time required to reasonably complete the Required Action.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

General

In addition to the ANSI N510 test requirements, and normal preventive and post-maintenance testing, there are a number of specific tests which must be performed to ensure proper functioning of the CRFA system.

SR 3.7.4.1

Standby systems should be checked periodically to ensure they will start and function properly. This Surveillance Requirement ensures each subsystem will start on demand and continue to operate. Operation with the heaters on for greater than or equal to 10 hours every 31 days reduces the buildup of moisture on the adsorbers and HEPA filters.

SR 3.7.4.2, SR 3.7.4.3, SR 3.7.4.4 and SR 3.7.4.5

These Surveillance Requirements demonstrate the designed filtration capability of the system is maintained by verifying the system flow rate, HEPA filters, and charcoal adsorbers satisfy the in-place testing acceptance criteria, the surveillance intervals and procedures of Reference 3 and ANSI N510-1975. The laboratory analysis of a representative carbon sample (SR 3.7.4.3) must be performed in accordance with the testing criteria of Regulatory Position C.6.a of Reference 3. The carbon sample to be used in this test must be obtained in accordance with Regulatory Position C.6.b of Reference 3. The in-place acceptance criteria of SR 3.7.4.2 and SR 3.7.4.3 are defined in Regulatory Position C.5 of Reference 3. The system flow rate is verified during subsystem operation for SR 3.7.4.2, SR 3.7.4.3 or SR 3.7.4.5 when tested in accordance with ANSI N510-1975.

In addition to the 18 month Frequency, SR 3.7.4.2, SR 3.7.4.3 and SR 3.7.4.4 are also required whenever major modifications or events (e.g. painting, fire or chemical release) which may have affected the integrity of the HEPA filters or charcoal adsorbers have occurred. For the purpose of this specification, "area being serviced by the filter" means the area where painting, fire or chemical release occurred and from which suction is being taken by an operating fan.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.4.6

This Surveillance Requirement verifies the duct heater performance of the CRFA system. The test is performed in accordance with Section 14 of ANSI N510-1975 with the exception of the 5% current phase balance of Section 14.2.3. The offsite power system for the Grand Gulf Nuclear Station consists of a non-transpositional 500-kV grid. The grid has an inherent unbalanced load distribution which results in unbalanced voltages in the plant. Voltage unbalances exceeding the 5% criteria of ANSI N510-1975 are not atypical.

SR 3.7.4.7

This Surveillance Requirement verifies the CRFA system will automatically switch to the isolation mode of operation to maintain control room habitability on receipt of an actuation signal.

Surveillance Frequencies

In general, Surveillance Frequencies are based on industry accepted practice and engineering judgement considering the unit conditions required to perform the test, the ease of performing the test and a likelihood of a change in the system/component status. The Surveillance Frequencies for testing of the HEPA filters and charcoal absorber units are consistent with the requirements of Reference 4.

REFERENCES

1. Grand Gulf FSAR, Section 6.5.1.1.

2. Grand Gulf FSAR, Section 15.

3. Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Post Accident Engineered Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants", Revision 2, March 1978.

4. Generic Letter 83-13, "Clarification of Surveillance Requirements for HEPA Filters and Charcoal Absorber Units in Standard Technical Specifications on ESF Cleanup Systems," March 2, 1983.

5. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

Grand Gulf Nuclear Station
Technical Specification Improvement Program

Revision Summary Sheet

Proposed LCO/Section: 3.7.5 Rev. 1 Main Condenser Offgas

Item	Change Description	Category
1	LCO 3.7.5 is reformatted from LIMITING CONDITION FOR OPERATION 3.11.2.7.	1
2	DELETED	
3	CONDITION A is reformatted from the ACTION statement.	1
4	CONDITION B revises the shutdown requirement of the ACTION statement to MSIV isolation.	3B
5	SR 3.7.5.1 is reformatted from SR 4.11.2.7.2 except as discussed below.	1
6	The Specification 4.0.4 exception in SR 4.11.2.7.2 (footnote "**" to page 3/4 11-17) is deleted as it is no longer applicable.	1
7	SR 4.11.2.7.1 is deleted. (LCO 3.3.7.12 is relocated). See Note 1 below.	4
8	The words "primary coolant" in SR 4.11.2.7.2.b is not included in SR 3.7.5.1.	4

NOTE

1. Per NRC letter on split document, the instrumentation is to be retained and also the GL on removal of RETs retains this requirement. SR 3.7.5.1 takes credit for the continuous monitoring of the offgas effluent in setting the surveillance frequency.

3.7 PLANT SYSTEMS

3.7.5 Main Condenser Offgas

LCO 3.7.5 The gross gamma radioactivity rate of the noble gases measured at offgas recombiner effluent shall be $\leq$ 380 millicuries/second, after 30 minutes decay.

APPLICABILITY: MODE 1,
MODES 2 and 3 with a steam jet air ejector in operation.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Gross gamma radioactivity rate > 380 millicuries/second after 30 minutes decay.	A.1 Restore gross gamma radioactivity rate to within limits.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Isolate all main steam lines.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Perform an isotopic analysis of a representative sample of gases taken at the offgas recombiner effluent.	31 days AND Once within 4 hours after a ≥ 50% increase in the nominal steady state fission gas release after factoring out increases due to changes in THERMAL POWER level

CROSS-REFERENCES: None

B 3.7 PLANT SYSTEMS

B 3.7.5 **Main Condenser Offgas**

BASES

BACKGROUND	During plant operation, steam from the low-pressure turbine is exhausted directly into the condenser. Air leakage and noncondensible gases are collected in the condenser, then exhausted through the steam-jet air ejectors to the Offgas System. The offgas from the main condenser normally includes radioactive gases. The Main Condenser Offgas System has been incorporated into the plant design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser and the water and condensibles are stripped out by the offgas condenser and moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the offgas condenser.
APPLICABLE SAFETY ANALYSES	The main condenser offgas gross gamma radioactivity rate is an initial condition of the Offgas System Failure Event (Ref. 1). The analysis assumes a gross failure in the Offgas System that results in the rupture of the Offgas System pressure boundary. The gross gamma radioactivity rate is controlled to ensure that during the event the calculated offsite doses will be well within the limits of 10 CFR 100. Main Condenser Offgas satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 2.
LCO	To ensure compliance with the assumptions of the Offgas System Failure Event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/MWt-sec at 30 minutes decay. The LCO is established consistent with this requirement (3833 MWt x 100 μCi/MWt-sec = 380 millicuries/second).

(continued)

BASES (continued)

APPLICABILITY

The LCO is applicable during MODES 1,2 and 3 when steam is being exhausted to the main condenser and steam jet air ejectors are being used to maintain condenser vaccum. In MODES 4 and 5, steam is not being exhausted to the main condenser and the requirements are not applicable.

ACTIONS

A.1

If the offgas radioactivity rate limit is exceeded, a limited time is permitted to restore the gross gamma radioactivity rate to within the limit because of the large margin to permissible dose and exposure limits.

B.1, B.2

If the gross gamma radioactivity rate is not restored to within the limits and the associated Completion Time is not met, all main steam lines must be isolated. This isolates the condenser from the source of the radioactive steam.

Completion Times

All Completion Times are based on industry accepted practice and engineering judgement considering the number of available systems and the time required to reasonably complete the Required Action.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This surveillance periodically analyzes a sample of the offgas to ensure that the required limits are satisfied. If the measured rate of radioactivity increases significantly (by $\geq$ 50% percent after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is performed within 4 hours, after the increase is noted, to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. The Surveillance Frequencies are considered adequate based on the availability of instrumentation to continuously monitor the offgas.

REFERENCES

1. Grand Gulf Unit 1 FSAR, Section 15.7.1.

2. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.