Entergy

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Attachment 2 contains proprietary information.

GNRO-2012/00125

October 26, 2012

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Final Feedwater Temperature Reduction License Amendment Request

Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

REFERENCE: NRC letter to Entergy Operations, Inc., *Grand Gulf Nuclear Station, Unit 1 – Issuance of Amendment Re: Extended Power Uprate (TAC No. ME4679)*, July 18, 2012 (ADAMS Accession #ML121210020)

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) proposes to revise the Grand Gulf Nuclear Station, Unit 1 (GGNS) Operating License (OL) to allow GGNS to operate with Final Feedwater Temperature Reduction (FFWTR) at the end of a fuel cycle (EOC) for the purpose of extending the cycle. The FFWTR flexibility would allow operating with a reduction of 100°F in feedwater temperature (from 420°F to 320°F) at rated power conditions. Specifically, this license amendment request (LAR) proposes to delete Section 2.C.(32) of the GGNS OL, which currently prohibits operating with partial feedwater heating for the purpose of extending the fuel cycle. The FFWTR LAR is provided in Attachment 1.

To support this LAR, analyses and evaluations have been prepared by Entergy and General Electric – Hitachi Nuclear Energy Americas, LLC (GEH). The results of the analyses and evaluations are documented in GEH Report NEDC-33671P, *Safety Analysis Report for Operation with Final Feedwater Temperature Reduction at Grand Gulf Nuclear Station*, Rev. 0, January 2012. NEDC-33671P is provided in Attachment 2.

GEH considers certain information provided in NEDC-33671P to be proprietary and, therefore, exempt from public disclosure pursuant to 10 CFR 2.390. Therefore, on behalf of GEH, Entergy requests the NRC withhold Attachment 2 from public disclosure in accordance with 10 CFR 2.390(b)(1). An affidavit for withholding information, executed by GEH, is provided in NEDC-33671P. A non-proprietary version of NEDC-33671P is provided in Attachment 3.

When Attachment 2 is removed from this letter, the entire document is NON-PROPRIETARY.

Attachment 4 contains the marked-up page of the GGNS OL that reflects the change proposed in Attachment 1.

Entergy has evaluated the FFWTR LAR in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and has determined that it involves no significant hazards consideration. The basis for this determination is included in Attachment 1.

This letter contains no new commitments.

If you have any questions or require additional information, please contact Guy Davant at 601-368-5756.

I declare under penalty of perjury that the foregoing is true and correct; executed on October 26, 2012.

Sincerely,

BSF/ghd

Attachments:
1. Final Feedwater Temperature Reduction License Amendment Request
2. GEH Report NEDC-33671P, Rev. 0, *Safety Analysis Report for Operation with Final Feedwater Temperature Reduction at Grand Gulf Nuclear Station* (Proprietary Version)
3. GEH Report NEDO-33671, Rev. 0, *Safety Analysis Report for Operation with Final Feedwater Temperature Reduction at Grand Gulf Nuclear Station* (Non-Proprietary Version)
4. Marked-Up Operating License Page

cc: Mr. Elmo E. Collins, Jr.
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ATTACHMENT 1

GRAND GULF NUCLEAR STATION

GNRO-2012/00125

FINAL FEEDWATER TEMPERATURE REDUCTION
LICENSE AMENDMENT REQUEST

FINAL FEEDWATER TEMPERATURE REDUCTION LICENSE AMENDMENT REQUEST

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) proposes to revise the Grand Gulf Nuclear Station, Unit 1 (GGNS) Operating License (OL) to allow GGNS to operate with Final Feedwater Temperature Reduction (FFWTR) at the end of the fuel cycle (EOC) for the purpose of extending the cycle. The FFWTR flexibility option would allow operating with a reduction of 100°F in the feedwater temperature (from 420°F to 320°F) at rated power conditions.

2.0 PROPOSED CHANGE

OL Section 2.C.(32), Partial Feedwater Heating, is affected by this LAR. The change is identified and discussed in Section 4.0, below. Attachment 4 contains the marked-up OL page indicating the proposed change.

3.0 BACKGROUND

As the end of the fuel cycle approaches, reactor thermal power gradually decreases from rated conditions if cycle operation continues. This condition commonly referred to as a power coastdown, results when core reactivity decreases below the level at which it can be maintained by withdrawing control rods and/or increasing reactor core flow. FFWTR provides a means to extend the operating cycle by inserting positive reactivity via reducing feedwater temperature, thus delaying the onset of the power coastdown period. Reducing feedwater temperature by 100°F at the end of an operating cycle could extend the cycle for weeks.

During Cycle 1, GGNS was analyzed to operate with feedwater heaters out of service (FWHOOS), which reduced feedwater temperature by 100°F from 420°F to 320°F at rated power during the normal operating cycle as documented in GGNS UFSAR Section 15B. As part of its recent extended power uprate[1], this FWHOOS capability was maintained. However, operating with partial or reduced feedwater heating for cycle extension (i.e., FFWTR) is currently prohibited by GGNS OL Section 2.C.(32). Section 2.C.(32) states:

> "Operation of the plant in the partial feedwater heating mode for the purpose of extending the normal fuel cycle shall be prohibited until analyses which justify that operation are provided to and approved by the NRC staff."

This OL condition was imposed in response to a requirement specified in Section 15.1, Abnormal Operational Occurrences, of NUREG-0831, *Safety Evaluation Report related to the operation of Grand Gulf Nuclear Station, Units 1 and 2,* Supplement #2, June 1982. Section 15.1 stated:

---

[1] GGNS' extended power uprate was approved by the NRC via NRC letter to Entergy Operations, Inc., *Grand Gulf Nuclear Station, Unit 1 – Issuance of Amendment Re: Extended Power Uprate (TAC No. ME4679)*, July 18, 2012 (ADAMS Accession #ML121210020).

> "The applicants were asked to justify that operation with partial feedwater heating to extend the cycle beyond the normal end-of-cycle condition would not result in a more limiting change in minimum critical power ratio than that obtained using the assumption of normal feedwater heating. The applicants indicated that analyses will be provided before operation in this mode, if a decision is made to operate in this mode. Until such analyses are provided, the staff will condition the license from operation in this mode."

This LAR provides to the NRC the analyses that justify such operation and removal of this restriction.

4.0 TECHNICAL ANALYSIS

FFWTR may be used to extend the operating fuel cycle with rated feedwater temperature reduction limited to 100°F. Analyses and evaluations supporting operating with FFWTR at GGNS have been prepared by Entergy and General Electric – Hitachi Nuclear Energy Americas, LLC (GEH). The results of the analyses and evaluations are documented in GEH Report NEDC-33671P, *Safety Analysis Report for Operation with Final Feedwater Temperature Reduction at Grand Gulf Nuclear Station*, which is provided in Attachment 2.

The NEDC-33671P analyses and evaluations determined that the effect of FFWTR on the following subjects is acceptable:

1) ECCS Performance
2) Containment System Performance
3) Reactor Asymmetric Loads
4) Reactor Coolant and Connected Systems
5) Anticipated Operational Occurrence (AOO) Performance
6) Anticipated Transient without Scram (ATWS) Mitigation Capability
7) Thermal-Hydraulic Stability Performance
8) High Energy Line Break
9) Feedwater Nozzle Fatigue, and
10) Low Power Scram Bypass Setpoint[2]

A detailed discussion of each subject is provided in NEDC-33671P.

The proposed OL change is identified and described below.

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2 For GGNS, the Low Power Scram Bypass Setpoint is the setpoint at which the Turbine Stop Valve Closure, Trip Oil Pressure – Low and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low trip functions are enabled.

4.1 Proposed Deletion of OL Section 2.C.(32), *Partial Feedwater Heating*

OL Section 2.C.(32) currently states:

"(32) Partial Feedwater Heating (Section 15.1, SER, SSER #2)

Operation of the plant in the partial feedwater heating mode for the purpose of extending the normal fuel cycle shall be prohibited until analyses which justify that operation are provided to and approved by the NRC staff."

Entergy proposes deleting Section 2.C.(32) based upon the justification provided in NEDC-33671P.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

Entergy has determined that the proposed change discussed in Section 4.0 does not require any exemptions or relief from regulatory requirements, other than the OL, and does not affect conformance with any General Design Criterion (GDC) differently than described in the Updated Final Safety Analysis Report (UFSAR).

5.2 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, Entergy Operations, Inc. (Entergy) requests an amendment to the facility Operating License (OL) No. NPF-29 for the Grand Gulf Nuclear Station (GGNS). This license amendment request proposes changes to the OL. The proposed change would allow GGNS to operate with partial feedwater heating for the purpose of extending the normal fuel cycle.

Entergy has evaluated the proposed license amendment request in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of GGNS in accordance with the proposed amendment presents no significant hazards. Entergy's evaluation against each of the criteria in 10 CFR 50.92 follows:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   RESPONSE: No.

   The effect of operating with Final Feedwater Temperature Reduction (FFWTR) on the probability and consequences of accidents, Anticipated Operational Occurrences (AOO), and events documented in the Updated Final Safety Analysis (UFSAR) was reviewed.

   The impact of FFWTR on the Design Basis Accident (DBA) Loss-of-Coolant Accident (LOCA) was considered. Evaluations and analyses determined that the current licensing basis peak cladding temperature (PCT) of the fuel remains applicable for operating the Grand Gulf Nuclear Station (GGNS) with FFWTR. The analysis results indicate the following:

- The calculated maximum fuel element cladding temperature does not exceed 2,200°F.
- The calculated total local oxidation does not exceed 17% times the total cladding thickness.
- The calculated total amount of hydrogen generated from a chemical reaction of the cladding with water or steam is less than 1% times the hypothetical amount if all the metal in the cladding cylinder were to react.
- The core remains amenable to long term cooling, and there is sufficient long term core cooling available.

Analysis also demonstrated that FFWTR operation at GGNS continues to meet design limits for the DBA-LOCA peak drywell pressure and temperature. Therefore, there is no increase in the consequence of an accident previously evaluated in the UFSAR.

The only AOO that requires consideration in assessing the effect of FFWTR on event consequences is the feedwater controller failure - increasing flow (FWCF). This is based upon the finding that the other AOOs are less sensitive to a reduction in feedwater temperature. The rated power and off-rated Power Distribution Limits, Critical Power Ratio (CPR), and Linear Heat Generation Rate (LHGR), for the FWCF event are validated on a cycle-specific basis to ensure compliance with: (1) the Safety Limit Minimum Critical Power Ratio (SLMCPR) and (2) the fuel rod thermal mechanical acceptance criteria of avoiding fuel centerline melt and 1% cladding plastic strain. Consequently, there is no increase in the consequences of an AOO previously evaluated.

The impact of FFWTR on the consequences of the following events was also considered: Anticipated Transient without Scram (ATWS), vessel overpressure, thermal-hydraulic stability, and High Energy Line Break (HELB). The evaluation of ATWS and vessel overpressure concluded the consequences of the events at normal feedwater temperature remain bounding for FFWTR. The evaluation of HELB determined the impact was bounded by the current design basis. The cycle-specific determinations and validations performed in accordance with NRC-approved methods ensure that the SLMCPR will be protected if a thermal-hydraulic instability event were to occur. Therefore, there is no increase in the consequence of these events previously evaluated in the UFSAR.

In addition, the following areas were also evaluated. The reactor power level and operating pressure are not changed. FFWTR has no effect on the decay heat. Current design limits associated with long-term containment analyses, including a recirculation suction line break (RSLB), loss of offsite power (LOOP), intermediate break accident (IBA), small break accident (SBA), and NUREG-0783 safety/relief valve (SRV) steam discharge events continue to be supported without change. Therefore, there is no increase in the consequence of these events previously evaluated in the UFSAR.

The probability of an accident is not affected by the proposed changes since no structures, systems or components (SSC) that could initiate an accident are affected. Therefore, the proposed changes do not significantly increase the probability of any previously evaluated accident.

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

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

RESPONSE: No.

The proposed change does not alter the design function of any SSC. The implementation of FFWTR operation does not create the possibility of a new or different kind of accident. Power Distribution Limits for CPR, LHGR and Average Planar Linear Heat Generation Rate (APLHGR), and OPRM setpoints, which are determined in accordance with NRC-approved methods and are included in the Core Operating Limits Report (COLR) as part of the normal reload licensing process, continue to assure that core operation is in accordance with the conditions currently assumed for event initiation.

FFWTR was reviewed against the accidents, AOOs, and events documented in the UFSAR. This review determined there is no adverse impact; the existing design basis remains bounding. In addition, the proposed change does not involve new system interactions or equipment modifications to the plant. FFWTR does not involve any new type of testing or maintenance. Therefore, there are no new design basis failure mechanisms, malfunctions, or accident initiators created by the proposed change.

The existing low power scram bypass setpoint based on turbine first stage pressure and the calculated change in steam flow was evaluated. The current setpoint is based on operating with a 100°F reduction in feedwater temperature; therefore, the setpoint is unaffected by FFWTR.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   RESPONSE: No.

   The AOOs and accidents described in the UFSAR were evaluated for effects caused by the reduced feedwater temperature. For cycle-independent considerations, the evaluations determined that the consequences of the events are either: (1) bounded by the current design and licensing basis results; (2) are within design acceptance criteria; or (3) do not change in a manner that would reduce the margin of safety. For cycle-specific considerations, cycle-specific analyses utilizing NRC-approved methods that produce the values of the limits documented in the COLR continue to assure that core operation is maintained within the existing design basis and safety limits. No design basis or safety limit is altered by the proposed change.

   The existing low power scram bypass setpoint based on turbine first stage pressure and the calculated change in steam flow was evaluated. The current setpoint is based on operating with a 100°F reduction in feedwater temperature; therefore, the setpoint is unaffected by FFWTR.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92(c), in that it:

(1) Does not involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Does not create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Does not involve a significant reduction in a margin of safety.

5.3 Environmental Considerations

Entergy has determined that the proposed amendment would not change a requirement with respect to installation or use of a facility or component located within the restricted area, as defined in 10 CFR 20, nor would it change an inspection or surveillance requirement. The proposed amendment:

(i) Does not involve a significant hazards consideration; or

(ii) Does not authorize a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite; or

(iii) Does not result in a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for a categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), Entergy concludes no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Entergy Operations, Inc. letter to the NRC (GNRO-2010/00056), *License Amendment Request - Extended Power Uprate*, September 8, 2010 (ADAMS Accession No. ML102660403)

2. NUREG-0831, *Safety Evaluation Report related to the operation of Grand Gulf Nuclear Station, Units 1 and 2,* Supplement #2, June 1982

3. NRC letter to Entergy Operations, Inc., *Grand Gulf Nuclear Station, Unit 1 – Issuance of Amendment Re: Extended Power Uprate (TAC No. ME4679)*, July 18, 2012 (ADAMS Accession #ML121210020)

ATTACHMENT 2

GRAND GULF NUCLEAR STATION

GNRO-2012/00125

<u>GEH REPORT NEDC-33671P</u>

<u>SAFETY ANALYSIS REPORT FOR OPERATION WITH FINAL FEEDWATER TEMPERATURE REDUCTION AT GRAND GULF NUCLEAR STATION</u>

(PROPRIETARY)

The header of each page in this attachment carries the notation "GEH Proprietary Information." The GEH proprietary information is identified by double square brackets. [[This sentence is an example. {3}]] The superscript notation {3} refers to Paragraph (3) of the accompanying affidavit contained in Attachment 3, which provides the basis for the proprietary determination. Specific information that is not so marked is not GEH proprietary.

ATTACHMENT 3

GRAND GULF NUCLEAR STATION

GNRO-2012/00125

GEH REPORT NEDO-33671

SAFETY ANALYSIS REPORT FOR OPERATION WITH FINAL FEEDWATER TEMPERATURE REDUCTION AT GRAND GULF NUCLEAR STATION

(NON-PROPRIETARY)

This is a non-proprietary version of Attachment 2 from which the proprietary information has been removed. The proprietary portions that have been removed are indicated by double square brackets as shown here: [[ ]].

GE Hitachi Nuclear Energy

NEDO-33671
Revision 0
DRF Section 0000-0141-4188 R2
October 2012

Non-Proprietary Information - Class I (Public)

SAFETY ANALYSIS REPORT FOR OPERATION WITH FINAL FEEDWATER TEMPERATURE REDUCTION AT GRAND GULF NUCLEAR STATION

INFORMATION NOTICE

This is a non-proprietary version of the document NEDC-33671P, Revision 0, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The design, engineering, and other information contained in this document is furnished for the purpose of supporting the Grand Gulf Nuclear Station license amendment request for Final Feedwater Temperature Reduction (FFWTR) in proceedings before the U. S. Nuclear Regulatory Commission. The only undertakings of GEH with respect to information in this document are contained in the contract between GEH and Entergy, and nothing contained in this document shall be construed as changing that contract. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, GEH makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

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EXECUTIVE SUMMARY

This report summarizes the analysis results to support the operation of Entergy Grand Gulf Nuclear Station (GGNS) with Final Feedwater Temperature Reduction (FFWTR). The safety and regulatory concerns addressed in this report include the Emergency Core Cooling System (ECCS) performance analysis for a Loss-of-Coolant Accident (LOCA), containment system response, reactor asymmetric loads, reactor coolant and connected systems, Anticipated Operational Occurrences (AOOs) performance, Anticipated Transient Without Scram (ATWS) mitigation capability, thermal-hydraulic stability, high energy line break (HELB), feedwater (FW) nozzle fatigue, and the low power scram bypass setpoint.

ACRONYMS

Term	Definition
ADS	Automatic Depressurization System
AOO	Anticipated Operational Occurrence
AP	Annulus Pressurization
ASDC	Alternate Shutdown Cooling
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BSP	Backup Stability Protection
BSW	Biological Shield Wall
CLTP	Current Licensed Thermal Power
CO	Condensation Oscillation
CPR	Critical Power Ratio (ΔCPR = Change in Critical Power Ratio)
CRGT	Control Rod Guide Tube
CUF	Cumulative Usage Factor
DBA	Design Basis Accident
DIVOM	Delta CPR over Initial MCPR Versus the Oscillation Magnitude
DP	Differential Pressure
DW	Drywell
ECCS	Emergency Core Cooling System
EOC	End-of-Cycle
EOC-RPT	End-of-Cycle Recirculation Pump Trip
EPU	Extended Power Uprate
EQ	Environmental Qualification
EQDW	ISLB and SSLB for EQ
ERCD	Early Reactor Cavity Drain
FCV	Flow Control Valve
FFWTR	Final Feedwater Temperature Reduction
FIL	Flow Induced Loads
FW	Feedwater

Term	Definition
FWCF	Feedwater Controller Failure – Increasing Flow
FWHOOS	Feedwater Heaters Out-of-Service
FWLB	Feedwater Line Break
GEH	GE-Hitachi Nuclear Energy Americas LLC
GGNS	Entergy Grand Gulf Nuclear Station
HELB	High Energy Line Break
HPCS	High Pressure Core Spray
IBA	Intermediate Break Accident
ICA	Interim Corrective Action
ICF	Increased Core Flow
ISLB	Intermediate Steam Line Break
JR	Jet Reaction
LOCA	Loss-of-Coolant Accident
LPCI	Low Pressure Coolant Injection
LPRM	Local Power Range Monitor
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
MELLLA	Maximum Extended Load Line Limit Analysis
MSLB	Main Steam Line Break
NFWT	Normal Feedwater Temperature
NRC	Nuclear Regulatory Commission
ODB	Original Design Basis
OFS	Orificed Fuel Support
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
OPRM	Oscillation Power Range Monitor
PBDA	Period Based Detection Algorithm
PCT	Peak Cladding Temperature
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary

Term	Definition
RDLB	Recirculation Discharge Line Break
RHR	Residual Heat Removal
RIPD	Reactor Internal Pressure Difference
RLB	Recirculation Line Break
RPV	Reactor Pressure Vessel
RSLB	Recirculation Suction Line Break
RWCU	Reactor Water Cleanup
SBA	Small Break Accident
SBYP	Steam Bypass
SC	Safety Communication
SER	Safety Evaluation Report
SLMCPR	Safety Limit Minimum Critical Power Ratio
SP	Suppression Pool
SRV	Safety Relief Valve
SRVT	SRV Discharge Events for NUREG-0783
SSC	Structure, System, and Component
SSLB	Small Steam Line Break
TFSP	Turbine First Stage Pressure
TT	Turbine Trip

1.0 INTRODUCTION

The objective of this report is to present the analytical results to support operation with Final Feedwater Temperature Reduction (FFWTR) at the end-of-cycle (EOC) to extend the normal fuel cycle at Grand Gulf Nuclear Station (GGNS). FFWTR is a flexibility option to the normal operating condition; therefore, the normal operating condition and the normal feedwater (FW) design temperature remain unchanged with the implementation of this flexibility option. The GGNS FFWTR flexibility option assumes a reduction of 100°F in the FW temperature at rated power conditions, which corresponds to a decrease from 420°F to 320°F. When operating with FFWTR, the FW temperature at rated power conditions shall not be less than 320°F, as this is the value that was used in the evaluation.

Feedwater Heaters Out-of-Service (FWHOOS) is a plant operating flexibility option allowing continued operation with less than the full FW system heating capacity available during the operating cycle. The effect of a FWHOOS 100°F temperature reduction was evaluated as part of the GGNS Extended Power Uprate (EPU) analyses (Reference 1). The results of those evaluations are also applicable to a FFWTR of 100°F.

The relationship between FWHOOS and FFWTR are summarized in the following table:

Flexibility Option	Exposure Range	Temperature Range	Reference
FWHOOS	Up to EOC	Up to 100°F reduction at rated power (i.e., 320°F to 420°F at 100% power)	Current GGNS license basis evaluated for EPU in Reference 1.
FFWTR	Beyond EOC	Up to 100°F reduction at rated power (i.e., 320°F to 420°F at 100% power)	Evaluated in this report.

The effect of FFWTR on the following subjects is addressed in this evaluation:

- Emergency Core Cooling System (ECCS) Performance Analysis
- Containment System Response
- Reactor Asymmetric Loads
- Reactor Coolant and Connected Systems
- Anticipated Operational Occurrence (AOO) Performance
- Anticipated Transient Without Scram (ATWS) Mitigation Capability
- Thermal-Hydraulic Stability
- High Energy Line Break (HELB)
- FW Nozzle Fatigue

- Low Power Scram Bypass Setpoint

1.1 Summary and Conclusions

The effect of a 100°F FW temperature reduction for a FWHOOS plant operating flexibility option was evaluated as part of the GGNS EPU evaluation in Reference 1 for an equilibrium core of GNF2 fuel and determined to be acceptable. The acceptability is consistent with operation at 100°F FFWTR. The sections of this report containing the summary of results for each subject are provided in Table 1-1.

1.2 Operating Conditions

1.2.1 Reactor Heat Balance

Table 1-2 summarizes heat balance conditions for normal feedwater temperature (NFWT) and FFWTR.

Figure 1-1 provides the heat balance condition for 420°F NFWT. Figure 1-2 indicates the heat balance conditions for a 100°F FFWTR.

1.3 Computer Codes

Nuclear Regulatory Commission (NRC)-approved or industry-accepted computer codes are used in the GGNS FFWTR evaluations. The primary computer codes used for the GGNS FFWTR evaluations are listed in Table 1-3. Exceptions to the use of the code or special conditions of the applicable safety evaluation report (SER) are included as notes to Table 1-3.

Table 1-1 Summary of Results Presented in this Report

Subject	Section	Result
ECCS Performance Analysis	2.0	Acceptable
Containment System Response	3.0	Acceptable
Reactor Asymmetric Loads	4.0	Acceptable
Reactor Coolant and Connected Systems	5.0	Acceptable
AOO Performance	6.1	Acceptable
ATWS Mitigation Capability	6.2	Acceptable
Thermal-Hydraulic Stability	6.3	Acceptable[1]
HELB	6.4	Acceptable
FW Nozzle Fatigue	6.5	Acceptable
Low Power Scram Bypass Setpoint	6.6	Acceptable

Notes:
1. The adequacy of the Option III Oscillation Power Range Monitor (OPRM) Trip-Enabled Region will be assessed for each reload cycle.

Table 1-2 Reactor Heat Balance for NFWT and FFWTR Conditions

Parameter	Unit	NFWT	FFWTR (-100°F)
Thermal Power	MWt / % Rated	4,408.0 / 100	4,408.0 / 100
Core Flow	Mlbm/hr / % Rated	112.5 / 100	112.5 / 100
Core Inlet Enthalpy	Btu/lbm	525.1	510.0
FW Temperature	°F	420.0	320.0
Dome Pressure	psia	1,040.0	1,020.0
Vessel Steam Flow	Mlbm/hr	18.968	16.730

Table 1-3 Computer Codes Used in the FFWTR Evaluations

Task	Computer Code*	Version or Revision	NRC Approved	Comments
Reactor Heat Balance	ISCOR	09	Y(1)	NEDE-24011-P Rev. 0 SER
Thermal-Hydraulic Stability	ISCOR PANAC ODYSY	09 11 05	Y(1) Y Y	NEDE-24011-P Rev. 0 SER NEDE-30130-P-A (7) NEDE-33213P-A
Reactor Internal Pressure Differences (RIPDs)	LAMB TRACG ISCOR	07 02 09	(3) Y(8) Y(1)	NEDE-20566P-A NEDE-32176P, Rev. 2 NEDC-32177P, Rev. 2 NRC TAC No. M90270 NEDE-24011-P Rev. 0 SER
Containment System Response	M3CPT LAMB	05 08	Y (3)	NEDO-10320, April 1971 (NUREG-0661) NEDE-20566P-A, September 1986
Annulus Pressurization (AP) Loads	TRACG GEAPL SAP4G SPECA ANSYS	04 01 07 05 11	N(10) N(6) N(6) N(6) N	NEDE-33440P, Rev. 2, March 2010 NEDE-25199, October 1979 NEDO-10909, Rev. 7, December 1979 NEDE-25181, Addendum 1, August 1996 (11)
ECCS-Loss-of-Coolant Accident (LOCA)	LAMB GESTR SAFER ISCOR TASC	08 08 04 09 03	Y Y Y Y(1) Y	NEDE-20566P-A NEDE-23785-1-PA Rev. 1 (4) (5) NEDE-24011-P Rev. 0 SER NEDC-32084P-A Rev. 2 (9)
Transient Analysis	PANAC ODYN	11 10	Y Y	NEDE-30130-P-A (2) NEDE-24154P-A

* The application of these codes to the analyses complies with the limitations, restrictions, and conditions specified in the approving NRC SER where applicable for each code.

(1) The ISCOR code is not approved by name. However, the SER supporting approval of NEDE-24011-P Rev. 0 by the May 12, 1978 letter from D. G. Eisenhut (NRC) to R. Gridley (GE) finds the models and methods acceptable, and mentions the use of a digital computer code. The referenced digital computer code is ISCOR. The use of ISCOR to provide core thermal-hydraulic information in RIPDs, Transient, ATWS, Stability, Reactor Core and Fuel Performance, and LOCA applications is consistent with the approved models and methods.

(2) The physics code PANACEA provides inputs to the transient code ODYN. The improvements to PANACEA that were documented in NEDE-30130-P-A were incorporated into ODYN by way of Amendment 11 of GESTAR II (NEDE-24011-P-A). The use of PANAC Version 11 in this application was initiated following approval of Amendment 26 of GESTAR II by letter from S.A. Richards (NRC) to G.A. Watford (GE) Subject: "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II Implementing Improved GE Steady-State Methods," (TAC NO. MA6481), November 10, 1999.

(3) The LAMB code is approved for use in ECCS-LOCA applications (NEDE-20566P-A), but no approving SER exists for the use of LAMB for the evaluation of RIPDs or containment system response. The use of LAMB for these applications is consistent with the model description of NEDE-20566P-A.

(4) "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants," NEDE-30996P-A, General Electric Company, October 1987.

(5) Letter, Richard E. Kingston (GEH) to NRC, "Transmittal of Revision 1 of NEDC-32950, Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," MFN 07-406, July 31, 2007.

(6) Not a safety analysis code that requires NRC approval. The code application is reviewed and approved by GE-Hitachi Nuclear Energy Americas LLC (GEH) for "Level-2" application and is part of GEH's standard design process. Also, the application of this code has been used in power uprate submittals.

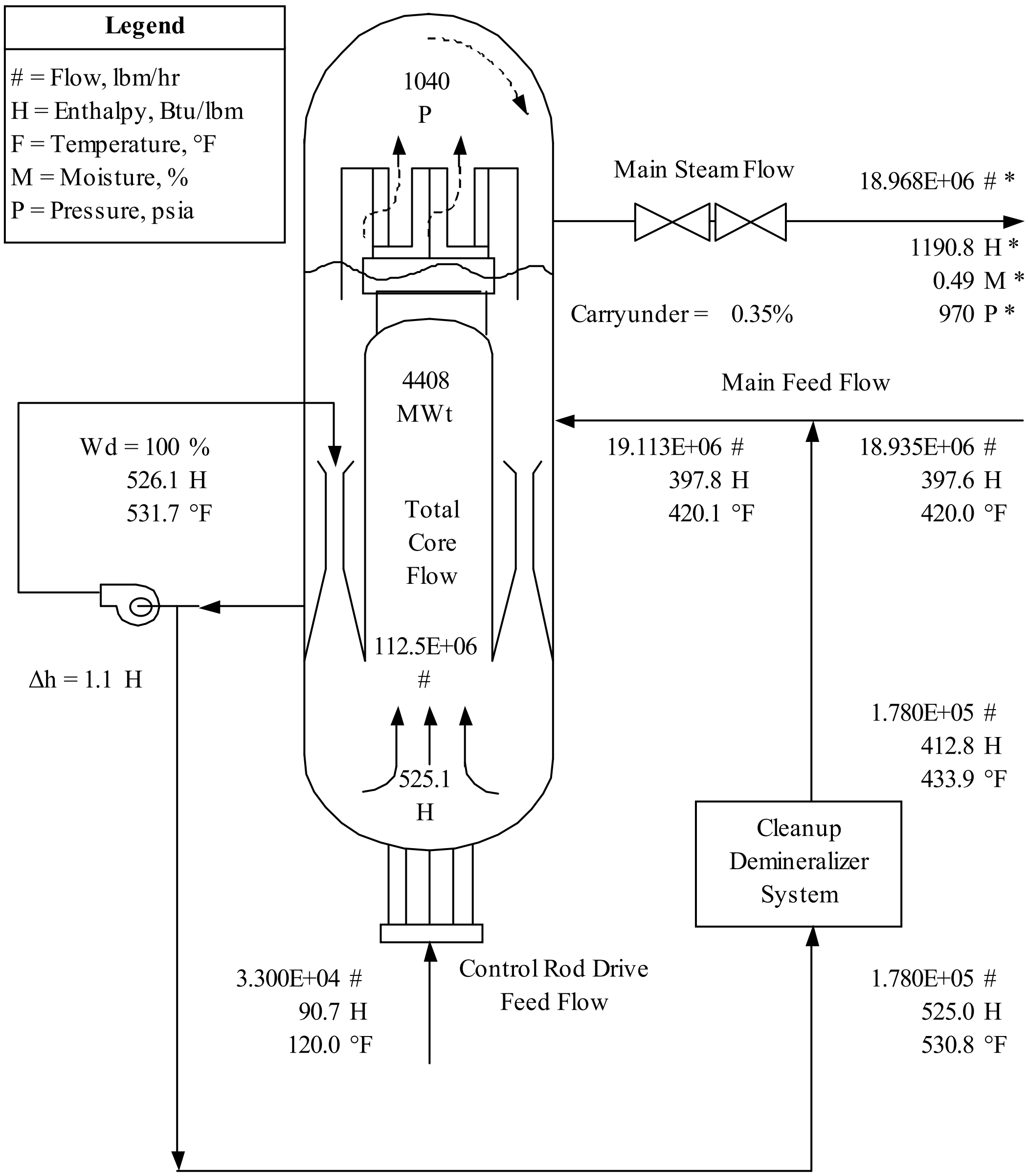
(7) The use of TGBLA Version 06 and PANACEA Version 11 was initiated following approval of Amendment 26 of GESTAR II by letter from S.A. Richards (NRC) to G.A. Watford (GE) Subject: "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II Implementing Improved GE Steady-State Methods," (TAC NO. MA6481), November 10, 1999.

(8) NRC has reviewed and accepted the TRACG application for the flow-induced loads on the core shroud as stated in NRC SER TAC No. M90270.

(9) The NRC approved the TASC-03A code by letter from S. A. Richards (NRC) to J. F. Klapproth (GE Nuclear Energy), Subject: "Review of NEDC-32084P, TASC-03A, A Computer Code for Transient Analysis of a Single Fuel Channel," TAC NO. MB0564, March 13, 2002.

(10) The application of TRACG04 for the calculation of AP loads has been described for ESBWR AP application in NEDE-33440P. The application of TRACG04 for the GGNS FFWTR has been applied in a manner consistent with NEDE-33440P.

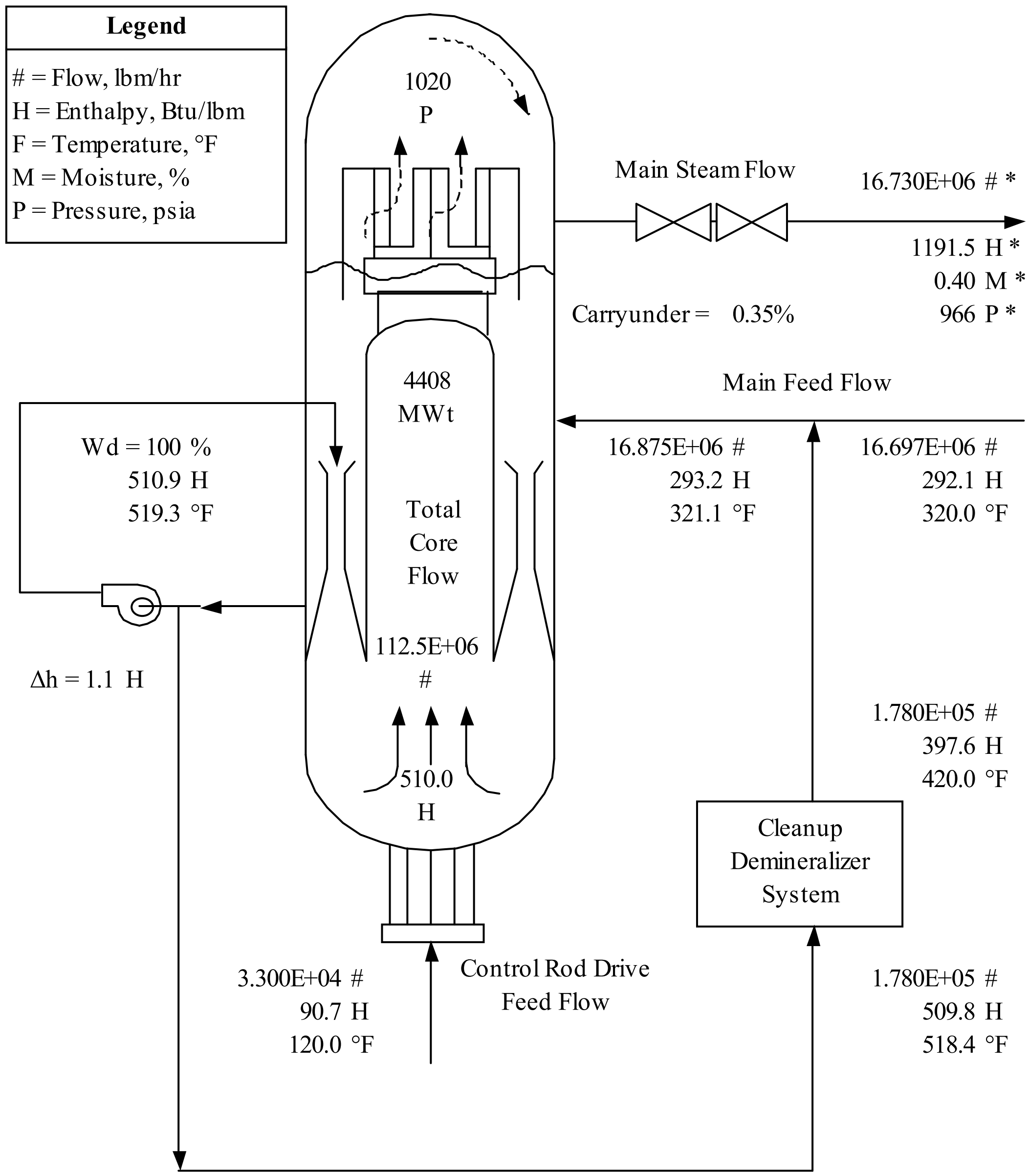
(11) ANSYS Finite Element Program, Service Pack 1, Level 2 certified installation in Wilmington, NC, ANSYS Inc., Canonsburg, PA.



*Conditions at upstream side of TSV

Core Thermal Power	4408.0
Pump Heating	11.2
Cleanup Losses	-5.9
Other System Losses	-1.1
Turbine Cycle Use	4412.2 MWt

Figure 1-1 Reactor Normal Feedwater Temperature (420°F)
4,408.0 MWt and 100% Flow Heat Balance



*Conditions at upstream side of TSV

Core Thermal Power	4408.0
Pump Heating	11.2
Cleanup Losses	-5.9
Other System Losses	-1.1
Turbine Cycle Use	4412.2 MWt

Figure 1-2 Reactor Feedwater Temperature Reduction (320°F) 4,408.0 MWt and 100% Flow Heat Balance

2.0 ECCS PERFORMANCE ANALYSIS

2.1 Analysis Approach

The effect of the 100°F FWHOOS condition on the ECCS performance for the limiting break failure combination was previously evaluated for the GGNS EPU (Reference 1) using the NRC-approved SAFER/GESTR-LOCA methodology documented in Reference 2. The results of this evaluation are also applicable to a 100°F FFWTR.

The limiting break and failure combination for GGNS is the design basis accident (DBA) recirculation suction line break (RSLB) under the limiting single failure of high pressure core spray (HPCS) – diesel generator indicated in Reference 1. The peak cladding temperature (PCT) trends from Reference 1 used to determine the limiting break and failure combination are not affected by FFWTR. The methodology documented in Reference 2 was used to analyze the FFWTR cases to determine the effect of FFWTR on the Licensing Basis PCT reported in Reference 1. The initial conditions for the FFWTR analysis are listed in Table 2-1.

2.2 Evaluation

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The limiting break and failure combination for GGNS was previously evaluated in Reference 1 for a 100°F FWHOOS temperature reduction for Appendix K assumptions using an approved set of ECCS parameters at the following power and flow conditions:

- 105% Original Licensed Thermal Power (OLTP) (4,105.5 MWt for Appendix K) and Maximum Extended Load Line Limit Analysis (MELLLA) core flow (90.9 Mlbm/hr), and
- Current Licensed Thermal Power (CLTP) (4,496.2 MWt for Appendix K) and MELLLA core flow (104.4 Mlbm/hr), and
- CLTP (4,496.2 MWt for Appendix K) and rated core flow (112.5 Mlbm/hr).

This evaluation is also applicable to a 100°F FFWTR.

Several power and flow conditions from the allowed operating domain were evaluated to demonstrate that a bounding PCT result is realized. Table 2-2 summarizes the PCT results for both NFWT and FWHOOS from Reference 1. Reference 3 indicates the restriction on the upper bound PCT is eliminated for all plants using the SAFER/GESTR-LOCA application methodology, which includes GGNS. In addition, Reference 1 indicates the 1,600°F restriction on the upper bound PCT is no longer applicable when evaluating the effect of changes and errors reported under the requirements of 10 CFR 50.46. Reference 1 indicates the licensing basis PCT is 1,690°F and bounds the PCT results for NFWT. The licensing basis PCT is determined based on a comparison of the nominal PCT results with PCT results considering Appendix K conservatisms. Additional uncertainties are applied statistically in the form of an adder to the difference. Given the PCT results shown in Table 2-2 for FFWTR, the current licensing basis PCT of 1,690°F would continue to be applicable and bound the cases with FFWTR as well as NFWT. Therefore, the 10 CFR 50.46 acceptance criteria continues to be met for up to a 100°F FFWTR because the PCT results remain below the licensing basis PCT.

The DBA-LOCA break result was calculated for FFWTR.

- The calculated maximum fuel element cladding temperature does not exceed 2,200°F.
- The calculated total local oxidation does not exceed 17% of the total cladding thickness.
- The calculated total amount of hydrogen generated from a chemical reaction of the cladding with water or steam is less than 1% of the hypothetical amount if all the metal in the cladding cylinder were to react.
- The core remains amenable to long term cooling.
- The sufficient long term core cooling remains available.

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The potential for rated power operation at slightly higher core exposures with FFWTR has no effect on the ECCS performance results because the PCT is based on a worst-case exposure of the peak bundle. The exposure-dependent Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits mitigate the core-wide effects and ensure that the fuel is not restricted by ECCS considerations at higher exposures up to the maximum planar exposure identified in the curve.

2.3 Conclusion

The results of the evaluation demonstrate that the criteria described in 10 CFR 50.46 and in the SAFER/GESTR SER (Reference 2) are met for FFWTR with GNF2 fuel.

Table 2-1 DBA-LOCA Initial Conditions for GGNS FFWTR

Plant Parameters	105% OLTP Appendix K	CLTP Appendix K
Thermal Power (MWt)	4,105.5	4,496.2
Thermal Power (% of 4,408.0 MWt)	93.1	102.0
Core Flow (Mlb/hr)	112.5	112.5
Core Flow (% of 112.5 Mlb/hr)	100.0	100.0
Vessel Steam Dome Pressure (psia)	1,100	1,100
FW Temperature (°F)	321.7	321.7

Table 2-2 SAFER/GESTR-LOCA Results Summary for the Limiting Recirculation Line Break

Power (% CLTP)	Flow (%)	Evaluation Assumption	Break Size	Location	Single Failure	FFWTR PCT (°F)	NFWT PCT (°F)
[[							
							]]

3.0 CONTAINMENT SYSTEM RESPONSE

3.1 Introduction

An evaluation was previously performed as part of the GGNS EPU (Reference 1) to determine the effect of operation with a 100°F FWHOOS temperature reduction on the design basis containment analysis performed for the GGNS EPU (Reference 1). This evaluation is also applicable to a FFWTR of 100°F.

[[

]] Therefore, the short-term containment response to a RSLB was analyzed to evaluate the effect of FFWTR. The containment responses of concern during this time period are:

- Peak drywell (DW) pressure
- Peak DW-to-wetwell pressure difference
- LOCA hydrodynamic loads

The short-term main steam line break (MSLB)-LOCA containment response and hydrodynamic loads analyses do not require review and evaluation for FFWTR operation [[
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The effect of FFWTR on the decay heat and vessel sensible energy is negligible because the reactor power level and operating pressure are not increased. [[

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3.2 Analysis Approach

For the GGNS EPU containment analysis (Reference 1), the M3CPT code (References 4 and 5) was used to evaluate the short-term pressure and temperature response to the DBA-LOCA. The analysis used the LAMB code (Reference 6) to calculate the blowdown flow rate and enthalpy, which are then used as input to M3CPT. Application of the LAMB blowdown model for containment analyses is identified in Reference 7.

3.3 Analysis Results

3.3.1 DBA-LOCA Short-Term Containment Pressure and Temperature

The short-term DBA-LOCA analysis was performed as part of the EPU analysis (Reference 1) using the M3CPT code with a 100°F FWHOOS temperature reduction and compared with NFWT cases.

As mentioned in Section 3.1, [[

]] Therefore, the pressure and temperature responses were evaluated only for the first 30-second period. The peak values of containment pressure and temperature responses during this period are compared to assess the effect of FFWTR on the containment system during the RSLB-LOCA.

That comparison showed that:

- [[

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The design limits for the DBA-LOCA peak DW pressure and temperature are 30 psig and 330°F, respectively. The GGNS LOCA peak DW pressure and temperature analysis of record was performed at CLTP and 102% of EPU power (100% = 4,408 MWt). The analysis demonstrated that FFWTR has a negligible effect on the EPU analysis and is bounded by the CLTP/EPU values of 27.0 psig and 330°F reported in Reference 1. The RSLB peak DW pressure of 23.9 psig and peak DW temperature of 252°F are within the design limits of 30 psig and 330°F, respectively.

3.3.2 DBA-LOCA Hydrodynamic Loads

Three types of hydrodynamic loads were addressed for the DBA-LOCA: (1) pool swell loads; (2) condensation oscillation (CO) loads; and (3) chugging loads. These are the loads considered in the existing design limits and for the FFWTR analysis values. The effect of FFWTR on these loads was evaluated as part of the GGNS EPU evaluation (Reference 1) by comparing the pressure and temperature responses obtained in Section 3.3.1 with those used in the load definitions for GGNS.

Pool Swell Loads

The pool swell loads are determined by the initial DW pressurization following the initiation of the DBA-LOCA. The DW pressure response used in the pool swell design load analysis for NFWT is presented at EPU conditions in Reference 1, which concluded that the pressurization exhibited by this pressure response bounds the initial DW pressurization for all cases. [[

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Condensation Oscillation Loads

CO loads result from oscillation of the steam-water interface that forms at the vent exit during the region of high vent steam mass flow rate. This occurs after pool swell and ends when the steam mass flux is reduced below a threshold value. CO loads increase with higher steam mass flux and higher suppression pool (SP) temperature. The CO loads include loads on submerged boundaries and submerged structures. [[

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Chugging Loads

Chugging loads include loads on the SP boundary and submerged structures and vent (downcomer) lateral loads. Chugging loads result from the collapse of steam bubbles that form at the vent exit. The chugging load definition for GGNS was based on the data from the chugging tests that covered thermal-hydraulic conditions expected for a Mark III containment geometry. Furthermore, chugging occurs when steam mass flux through the vent is not high enough to maintain a steady steam/water interface at the vent exit. Consequently, chugging occurs at the tail end of the DBA-LOCA or intermediate break accident (IBA), or during a small break accident (SBA) with the reactor at pressure. [[

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3.4 Conclusion

Based on the evaluations presented in this section, it is concluded that FFWTR operation at GGNS has no adverse effect on the DBA-LOCA containment pressure and temperature response. It is also concluded that the current LOCA hydrodynamic loads definition for GGNS is not affected by FFWTR operation.

4.0 REACTOR ASYMMETRIC LOADS

The reactor asymmetric loads analysis for EPU (Reference 1) assumed a FWHOOS 100°F temperature reduction. This analysis is also applicable to a 100°F FFWTR.

The Original Design Basis (ODB) mass and energy release rate profiles used in developing the asymmetric loads were calculated using the methods from NEDO-24548, "Annulus Pressurization Load Adequacy Evaluation" (Reference 8). Due to GEH's Safety Communication (SC) 09-01 (Reference 9), the large pipe break mass and energy release rates and AP time histories at EPU conditions were recalculated using TRACG (References 10 and 11).

Because of the issues identified in SC 09-01, the simplistic instantaneous break mass and energy release methodology (Reference 8) could potentially result in shifts of the frequency content of the AP response away from the resonant frequencies of the structures and components which could underestimate the dynamic amplification of the pressurization loads. However, NEDO-24548 methodology has not been shown to be non-conservative in any analysis performed to date (Reference 12). Evaluations were performed for EPU conditions (including a FWHOOS 100°F temperature reduction) to determine the effect of the AP load methodology change and EPU operation on the dynamic structural response of the reactor pressure vessel (RPV), reactor internals, piping and containment structures.

The following line breaks in the annulus region (between RPV and biological shield wall (BSW)) were evaluated for the effects of FFWTR:

- RSLB;
- Recirculation Discharge Line Break (RDLB); and
- Feedwater Line Break (FWLB).

These are the same pipe breaks considered in the original design basis. The plant design and licensing basis events for AP loads are the RSLB, RDLB, and the FWLB events. All three events were evaluated for FFWTR conditions consistent with SC 09-01. Other line breaks are outside of the existing GGNS design and licensing basis.

Analyses were performed for the large piping segment breaks within the annulus for effects including the structural dynamic response of the reactor vessel, reactor vessel internals, attachments to the vessel, and attachments to the BSW. Conditions analyzed include EPU conditions, several power flow points along the MELLLA, and the increased core flow (ICF) boundary from minimum core flow to maximum core flow through maximum EPU power. The breaks analyzed include the reactor recirculation discharge, reactor recirculation suction, and FW lines.

Results of the mass and energy release analysis were used in the AP analysis to determine the time-dependent AP response profiles.

The results of the EPU analysis show that the containment structures, systems, and components (SSCs) important to safety will continue to be protected from the dynamic effects resulting from pipe breaks and that the subcompartments will continue to have sufficient margins to prevent fracture of the structure due to pressure differences across the walls following implementation of the proposed EPU.

5.0 REACTOR COOLANT AND CONNECTED SYSTEMS

5.1 Reactor Internal Pressure Differences

FFWTR has a potential effect on RIPD analysis because lower steam generation in the core could change RPV depressurization and could change the flow mismatch between the steam generated in the core and the steam leaving the RPV through the break. Operation in the ICF domain results in higher initial core flow relative to rated core flow conditions and therefore yields a higher pressure difference across the components. ICF bounds operation at lower core flows such as those for the MELLLA domain.

The effect of a FWHOOS 100°F FW temperature reduction on RIPDs was previously evaluated as part of the GGNS EPU analysis (Reference 1). The results of that evaluation are also applicable to a FFWTR of 100°F.

5.1.1 RIPD Analysis Approach and Inputs

The RIPD analysis (including fuel lift margin and control rod guide tube (CRGT) lift force) was performed for the Faulted condition. The RIPDs at the Emergency condition are bounded by those at the Faulted condition due to a slower depressurization rate, with one exception. The pressure difference for the core plate at the Emergency condition is 1.0 psid higher than the pressure difference at the Faulted condition due to a later core flow surge after initial depressurization as a result of core flashing. The RIPDs at both Normal and Upset conditions with NFWT bound those with FFWTR due to lower steam generation.

The Faulted condition RIPD values in Reference 1 were calculated using the LAMB model to analyze the limiting MSLB inside containment accident.

5.1.2 RIPD Analysis Results

The results of the RIPD calculation with FFWTR up to 100°F from Reference 1 are shown in Tables 5-1 through 5-3.

The RIPD results presented herein were used for further structural integrity evaluation.

5.2 Acoustic and Flow-Induced Loads

Acoustic and flow-induced loads on the jet pump, core shroud and shroud support due to RSLB were evaluated as part of the GGNS EPU evaluation in Reference 1 because FFWTR would increase downcomer subcooling in the limiting MELLLA domain and consequently increase the loads on these components.

5.2.1 Analysis Approach and Inputs

The following assumptions and initial conditions were used in the determination of the acoustic and flow induced loads for a GGNS FFWTR of 100°F.

Analytical Assumptions [1]	Bases/Justifications
102 P / 100 F, NFWT	Consistent with GGNS Rated Licensing Basis
102 P / 100 F, FFWTR 102 P / 92.8 F, FFWTR	MELLLA Power/Flow State Point at Full Power
54.99 P / 34.0 F, FFWTR	MELLLA Upper Boundary at Low Recirculation Pump Speed with Maximum Flow Control Valve (FCV) Position

Note:

1. 2% additional power is assumed for accident application, consistent with Reference 1.

5.2.2 Analysis Results

The baseline flow-induced loads (FILs) at 90.2% of CLTP (3,976 MWt) / 100% of core flow are summarized in Table 5-4, and the FIL multipliers for the off-rated conditions are shown in Table 5-5. [[

]] The acoustic load is summarized in Table 5-6. The acoustic and flow-induced loads were used for the structural evaluation in Section 5.3.

5.3 Reactor Internals Structural Integrity

The structural integrity evaluation of the reactor internals with a FWHOOS 100°F FW temperature reduction was previously evaluated as part of the GGNS EPU analysis (Reference 1). The results of that evaluation are also applicable to a FFWTR of 100°F.

The loads considered in the EPU evaluation of the RPV internals in Reference 1 included RIPDs, deadweight, seismic loads, hydrodynamic loads such as SRV, LOCA, AP / Jet Reaction (JR) loads, acoustic loads, FILs due to recirculation line break (RLB), fuel lift loads, flow loads and thermal loads.

The EPU evaluation was performed consistent with the load combinations and American Society of Mechanical Engineers (ASME) Code allowable stresses and other acceptance criteria considered in the design basis (EPU) evaluation (Reference 1). The design basis evaluation was performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, 1974 Edition with Addenda to and including the Summer 1976 Addenda.

The results of the evaluation were calculated based on the limiting load combination, including flow-induced and acoustic loads. The acoustic load was used in the evaluation because the FIL is bounded by the acoustic load and the loads do not exist simultaneously. Flow-induced and acoustic loads were not used in the fatigue assessment because they are Faulted condition loads. A fatigue assessment is performed only for the Normal/Upset (service level A/B) condition.

The following reactor internal components were evaluated as part of the EPU analysis in Reference 1:

Core Support Structure Components

- Shroud Support
- Shroud
- Core Plate
- Top Guide/Grid
- Control Rod Drive Housing
- Control Rod Guide Tube
- Orificed Fuel Support (OFS)
- Peripheral Fuel Support

Non-Core Support Structure Components

- Fuel Channel
- Steam Dryer
- FW Sparger
- Jet Pump Assembly
- Core Spray Line and Sparger
- Access Hole Cover
- Shroud Head and Steam Separator Assembly
- In-Core Housing and Guide Tube
- Vessel Head Cooling Spray Nozzle
- Low Pressure Coolant Injection (LPCI) Coupling

FFWTR has no effect on the seismic, SRV, LOCA, and AP/JR loads; hence, the current design basis seismic, SRV, LOCA, and AP/JR loads remain valid for the reactor internals.

All stresses and fatigue usage factors are within the design basis ASME Code allowable limits, and the RPV internal components are demonstrated to be structurally qualified for operation at EPU conditions.

For Normal, Upset, Emergency and Faulted conditions, the EPU evaluation in Reference 1 remains bounding for the reactor internals because all applicable loads for FFWTR are either bounded by those considered in the EPU evaluation or have a negligible effect on the structural integrity of the reactor internals. Therefore, the reactor internal components remain qualified for FFWTR.

5.4 Reactor Coolant Pressure Boundary Piping

The reactor coolant pressure boundary (RCPB) piping, supports, and restraints were previously evaluated at EPU conditions (Reference 1). The evaluation assumed a FWHOOS 100°F temperature reduction; therefore, this analysis is also applicable for a 100°F FFWTR. The results of the EPU evaluation in Reference 1 showed that the change in the stresses on the piping, supports, and restraints will continue to meet the applicable ASME Code requirements.

Table 5-1 Summary of RIPD Results (Faulted Conditions)

Components	Faulted 37.25 P / 105 F [1] FFWTR (psid)	Faulted 102 P / 105 F [2] FFWTR (psid)	Emergency 102 P / 105 F [2] FFWTR (psid)
Shroud Support Ring and Lower Shroud	[[		
Core Plate and Guide Tube			
Upper Shroud			
Shroud Head			
Shroud Head to Water Level, Irreversible			
Shroud Head to Water Level, Elevation			
Channel Wall – Core Average Power Bundle			
Channel Wall – Maximum Power Bundle			
Top Guide			
Steam Dryer			]]

Notes:

1. 37.25 P = 37.25% of CLTP = 1,642.0 MWt; 105 F = 105% of Core Flow = 118.12 Mlbm/hr
2. 102 P = 102% of CLTP = 4,496.2 MWt; 105 F = 105% of Core Flow = 118.12 Mlbm/hr
3. Bounded by the hot standby condition in Reference 1, which does not change with FFWTR.

Table 5-2 Minimum Fuel Lift Margin Analysis Results

Case Conditions	Average Power Bundle (lbf)	Hot Power Bundle (lbf)
102 P / 105 F FFWTR	[[	
37.25 P / 105 F FFWTR		]]

Table 5-3 Maximum CRGT Lift Forces

Case Conditions	Average Power Bundle (lbf)	Hot Power Bundle (lbf)
102 P / 105 F FFWTR	[[	
37.25 P / 105 F FFWTR		]]

Table 5-4 Summary of Baseline Flow-Induced Loads Results

Item	Component	Parameter	Unit	Baseline Loads[1]
1	Shroud	Baseline Force	kips	[[
2		Baseline Moment at the Shroud Centerline	10^6 in-lbf	
3	Jet Pump	Baseline Force	kips	
4		Baseline Moment at the Jet Pump Centerline	10^6 in-lbf	]]

Note:

1. Loads at rated condition of 90.2 P (3,976 MWt) / 100 F with NFWT.

Table 5-5 Summary of Flow-Induced Load Multipliers

Item	Component	Operating Conditions	Load Multiplier[1]
1	Shroud/Jet Pump	NFWT 90.2 P / 100 F	[[
2		FFWTR 102 P / 92.8 F MELLLA Point	
3		FFWTR 54.99 P / 34.0 F MPS	
4		FFWTR 102 P / 100 F	]]

Note:

1. For off-rated conditions, the multipliers shall be applied for the baseline loads in Table 5-4.

Table 5-6 Summary of Acoustic Loads Results

Item	Component	Parameter	Unit	Maximum Loads[1]
1	Shroud	Total Lateral Force	kips	[[
2		Moment at the Base of the Shroud Centerline	10^6 in-lbf	
3	Shroud Support	Total Vertical Force	kips	
4		Moment at the Shroud Support Plate Outside Edge Nearest the Break	10^6 in-lbf	
5	Jet Pump	Total Lateral Force	kips	
6		Moment at the Center of the Base of the Jet Pump	10^6 in-lbf	]]

Note:

1. The loads are applicable for all conditions.

6.0 OTHER RELATED TECHNICAL ISSUES

6.1 AOO Performance

AOO events were previously evaluated at EPU conditions (Reference 1). The evaluation assumed a FWHOOS 100°F temperature reduction; therefore, this analysis is also applicable for a 100°F FFWTR.

The only AOO that requires consideration in assessing the effect of FFWTR on operating limits is the feedwater controller failure – increasing flow (FWCF). This is based upon the finding that the other AOOs are less sensitive to a reduction in FW temperature than FWCF, which is affected by the increase in core inlet subcooling prior to the turbine trip (TT) on high water level. Any effects of a slightly higher core exposure with FFWTR on the AOO performance is addressed as part of the cycle-specific reload analysis.

The FWCF EPU evaluation with FFWTR confirmed FWCF to be non-limiting at rated power conditions.

At off-rated power conditions, the subcooling portion of the FWCF event increases the severity of the transient response relative to other pressurization events. The FWCF event was evaluated with FFWTR at off-rated power conditions and the results confirm no change to the GGNS Cycle 19 power dependent limits.

6.2 ATWS Mitigation Capability

The effect of FFWTR on ATWS performance has been previously evaluated on a generic basis. These evaluations have shown that peak values for fuel surface heat flux, vessel bottom pressure, and SP temperature were all reduced when the FW temperature was reduced. This improvement in the ATWS performance was attributed to lower initial steam flow conditions resulting from the lower FW temperature. This condition causes a reduction in the vessel pressurization rate and in the mass/energy released into the wetwell.

As a result of the FFWTR, the steam generation rate and core void fraction are reduced. The lower steam generation rate is produced because more of the core heat is needed to heat up the colder moderator in the core. The lower steam generation rate increases the ratio of steam flow rate through the relief valves to steam generation rate, and therefore, the peak vessel pressure is lower. There is also less steam released to the SP so the pool heats up less.

The initial conditions associated with FFWTR result in a milder transient response relative to the achieved peak and integrated power. Additionally, the PCT occurs in the short-term part of an ATWS event where the FFWTR promotes improved heat transfer. The resulting PCT during the ATWS event with FFWTR is bounded by that predicted with NFWT. The effect of slightly higher core exposures associated with FFWTR on the exposure-dependent core parameters is bounded by the values applied in the current ATWS analysis.

Therefore, it is concluded, consistent with Reference 1, that the ATWS analysis results for GGNS at NFWT will bound that for FFWTR conditions.

6.3 Thermal-Hydraulic Stability

6.3.1 Option III Solution

GGNS has implemented stability solution Option III (Reference 13). Option III is a detect and suppress solution that combines closely spaced Local Power Range Monitor (LPRM) detectors into "cells" to effectively detect either core-wide or regional modes of reactor instability. These cells are termed OPRM cells and are configured to provide local area coverage with multiple channels. Plants implementing Option III have installed hardware to combine the LPRM signals and to evaluate the cell signals with instability detection algorithms. Of these algorithms, only the Period Based Detection Algorithm (PBDA) is credited in the Option III licensing basis. This algorithm provides an instrument setpoint designed to trip the reactor before an oscillation can grow to the point where the Safety Limit Minimum Critical Power Ratio (SLMCPR) is violated.

The current stability reload licensing basis is to calculate the limiting Operating Limit Minimum Critical Power Ratio (OLMCPR) required to protect the SLMCPR for both steady-state and transient stability events as specified in the Option III methodology (Reference 14). These stability-based OLMCPR values are calculated for a range of possible OPRM amplitude setpoints. Selection of appropriate instrument setpoints can then be made based upon the actual OLMCPR to provide adequate SLMCPR protection. As part of the GGNS EPU implementation, the setpoint development included the OPRM penalties discussed in Reference 15.

The Option III stability-based OLMCPR calculation requires the use of the regional Delta CPR over Initial Minimum Critical Power Ratio (MCPR) Versus the Oscillation Magnitude (DIVOM) curve per Reference 14. It was concluded in Appendix B of Reference 14 that variations in FW temperature (and therefore inlet subcooling) have very little effect on the slope of the DIVOM curve. The MCPR change during the flow reduction prior to the start of instability is also not sensitive to the reduction in FW temperature. Such small effects on the MCPR change prior to the onset of instability and DIVOM are bounded by the conservatisms in the method. Therefore, the stability-based OLMCPR values, including the penalties discussed in Reference 15, will not change due to FFWTR.

6.3.2 Backup Stability Protection

GGNS implements Backup Stability Protection (BSP) should the Option III OPRM system be declared inoperable (Reference 16). The BSP regions consist of two regions: Region I (Scram) and Region II (Controlled Entry). In addition, a minimum region size is determined from the Base BSP regions as described in Reference 16. If a calculated BSP region state point is located inside the corresponding Base BSP region state point, the calculated BSP region state point is replaced by the corresponding Base BSP region's state point (denoted by appending "-ICA" to the label in Figure 6-1). The Modified Shape Function is used to draw the BSP regions as approved in Reference 17. BSP regions are determined for both NFWT and FFTWR on a reload-specific basis. The BSP regions are also used to confirm, on a reload-specific basis, that the OPRM Trip-Enabled Region is adequate.

Demonstration BSP analyses were performed for 100°F FW temperature reduction, all other conditions being the same as in the EPU evaluation (Reference 1). The resulting BSP regions are shown in Figure 6-1. Figure 6-1 also shows that the OPRM Trip-Enabled Region envelopes the BSP regions; therefore, it is adequate.

6.3.3 Conclusion

FFWTR operation complies with the current licensing requirements for the Option III stability solution. The Option III solution is fully capable of supporting FFWTR, because it has been demonstrated that FFWTR does not adversely affect the core MCPR performance during an instability event. The adequacy of the Option III OPRM Trip-Enabled Region will be assessed for each fuel cycle reload. The effect of the higher core exposure with FFWTR on the stability analysis results is addressed as part of the cycle specific reload analyses because the reload stability analyses are performed to include the extended EOC exposure. If the Option III OPRM system is declared inoperable, the cycle-specific BSP regions are fully capable of supporting FFWTR operation, as shown by the demonstration analysis in Figure 6-1.

6.4 High Energy Line Break

The following HELBs were evaluated for the effects of FWHOOS as part of the EPU (Reference 1) evaluation, and the results of these analyses are also applicable to FFWTR:

- MSLB in the Main Steam Tunnel
- FWLB in the Main Steam Tunnel
- Residual Heat Removal (RHR) Line Break
- Reactor Core Isolation Cooling (RCIC) Steam Line Break
- Reactor Water Cleanup (RWCU) Line Break

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]] Therefore, FFWTR has no effect on the mass and energy releases from a FWLB in the main steam tunnel. Because the primary source feeding the RHR line break is FW, FFWTR also has no effect on the mass and energy releases from the RHR line break.

FFWTR does affect RWCU postulated line breaks [[

]] The effect of FFWTR was evaluated for five postulated RWCU break locations: RWCU pump room, RWCU heat exchanger room, filter/demineralizer room, holding pump room, and valve nest room.

The RWCU mass and energy releases for FFWTR, which may affect subcompartment pressures and temperatures, were bounded by the current RWCU mass and energy releases.

6.5 Feedwater Nozzle Stress and Fatigue

The fatigue experienced by the FW nozzle results from two phenomena: system cycling (including dynamic cycling) and rapid cycling. System cycling is caused by major temperature changes associated with system transients. Dynamic cycling, due to mechanical loading during scram events, was calculated separately from other system cycling cumulative usage factor (CUF) values in the evaluation. Rapid cycling is caused by small, high frequency temperature fluctuations resulting from the mixing of relatively colder nozzle annulus water with hot reactor water. The colder water impinging the nozzle originates from leakage past the thermal sleeve and from the boundary layer of colder water formed by heat transfer through the thermal sleeve.

Any system cycling stress or fatigue due to FFWTR conditions would be bounded by the "reduction to 0% power" (from rated power) transient. The reduction to 0% power transient is a reactor power reduction from rated power conditions to 0% power prior to a complete shutdown. It consists of a 155°F FW temperature reduction which bounds the 100°F FFWTR, and it occurs more rapidly than FFWTR. Therefore implementing FFWTR would reduce the effect of this transient with respect to stress and system cycling. The dynamic cycling is unaffected by the changes in FW temperature. .

The number of design cycles for the reduction to 0% power transient is 111 cycles. As long as the total number of reduction to 0% power (with or without implementing FFWTR) transients does not exceed 111 cycles, the system cycling CUF results previously evaluated are unaffected/bounding.

The number of design cycles does not represent a design limit. The fatigue for a component is normally the result of several different thermal and pressure transients. Exceeding the number of cycles for one transient does not necessarily imply the fatigue usage will exceed an acceptance limit. The fatigue monitoring program will monitor all necessary plant transients to ensure the fatigue usage remains less than the allowable limit. In the event that the monitored usage factor is predicated to exceed the allowable value for any component, appropriate corrective action will be taken in accordance with the corrective action program.

FFWTR operation affects the rapid cycling fatigue usage. This is primarily for two reasons. First, the transient temperature swing and rate of change associated with the mode of operation is relatively small and thus does not affect the system cycling usage factor. As a consequence, different transient behavior caused as a result of FFWTR operation does not have an effect on system cycling fatigue usage. Second, the time spent at reduced FW temperature is a finite contributor to rapid cycling fatigue usage. Because of this, the lower FW temperatures experienced during FFWTR operation could have an adverse effect on rapid cycling fatigue.

The stress and system plus dynamic cycling fatigue usage was evaluated in Reference 1. There is no increase in fatigue usage from system or dynamic cycling for the FW nozzles and FW piping at GGNS due to the addition of FFWTR. These conditions are bounded by other events used in the stress and system plus dynamic cycling fatigue analyses.

For the rapid cycling analysis, the ASME Boiler and Pressure Vessel Code 2001 Edition was used for the instantaneous coefficient of thermal expansion for carbon steel as a function of temperature. Because the values from the 2001 Edition of the Code are larger than those from the code year of the CLTP analysis (ASME Boiler and Pressure Vessel Code 1974 Edition with Addenda to and including Summer 1976), and because the alternating stress is proportional to this coefficient, it is conservative to use the newer Code.

A rapid cycling fatigue usage calculation for FWHOOS (100°F reduction for 313 days) was performed in Reference 1. FFWTR conditions are the same as for FWHOOS, and the time allotted for the two flexibility options together is the same as the analyzed-for time allotted to FWHOOS alone. Therefore the rapid cycling evaluation considering EPU with FWHOOS bounds the rapid cycling evaluation considering EPU with FFWTR.

Table 6-1 provides the projected 40-year usage factors and the allowable CUF for the FW safe end and the nozzle forging. In addition, the fatigue monitoring program will ensure that the actual fatigue usage will remain below the allowable value.

In summary, the rapid cycling fatigue usage added to the corresponding system plus dynamic cycling fatigue usage results in a maximum total 40-year cumulative fatigue usage factor of 0.581, which is less than the ASME Code allowable value of 1.0, and the stress evaluation results do not change.

6.6 Reactor Heat Balance for Low Power Scram Bypass Setpoint Assessment

The low power scram bypass function is used to reduce scrams and recirculation pump trips at low power levels where the turbine steam bypass system is effective for mitigating TTs and generator load rejections. This function bypasses the End-of-Cycle Recirculation Pump Trip (EOC-RPT) and scram on fast closure of the Turbine Stop or Control Valves at low power levels. The low power scram bypass setpoint is defined in terms of percent rated thermal power; the power level input is derived based on a Turbine First Stage Pressure (TFSP) correlation.

The GGNS Technical Specifications include a number of power-dependent setpoints that use TFSP as an indication of core power. These setpoints include: (i) Rod Withdrawal Limiter High Power Setpoint; (ii) TFSP Scram Bypass Permissive and EOC-RPT Bypass Permissive; and (iii) Rod Pattern Controller Low Power Setpoint. Plant operation with FFWTR affects the TFSP due to the reduced steam generation. These setpoints are currently based on operation with 100°F FW temperature reduction. Consequently, these setpoints are unaffected by FFWTR.

TFSP is affected by the lower steam generation when the plant is operated with reduced FW temperatures, as with FWHOOS or FFWTR. However, the low power scram bypass setpoint was modified as part of the EPU based on operation with FWHOOS with a 100°F reduction in the rated FW temperature. Because extending the cycle with FFWTR will not further affect the TFSP and the maximum temperature reduction is unchanged, this setpoint is not affected by the FFWTR.

Table 6-1 Feedwater Nozzle Projected 40-Year Usage Factors for CLTP with FFWTR/FWHOOS

Location	40-Year Projected Usage Factor	Allowable
Safe End – Stainless Steel	$0.293_{(s+d)} + 0.157_{(r)} = 0.450_{(t)}$	1.0
Safe End – Carbon Steel	$0.538_{(s+d)} + 0.008_{(r)} = 0.546_{(t)}$	1.0
Nozzle Forging – Low Alloy Steel	$0.564_{(s+d)} + 0.017_{(r)} = 0.581_{(t)}$	1.0

Notes: (s+d) = system plus dynamic CUF
(r) = rapid cycling CUF
(t) = total CUF

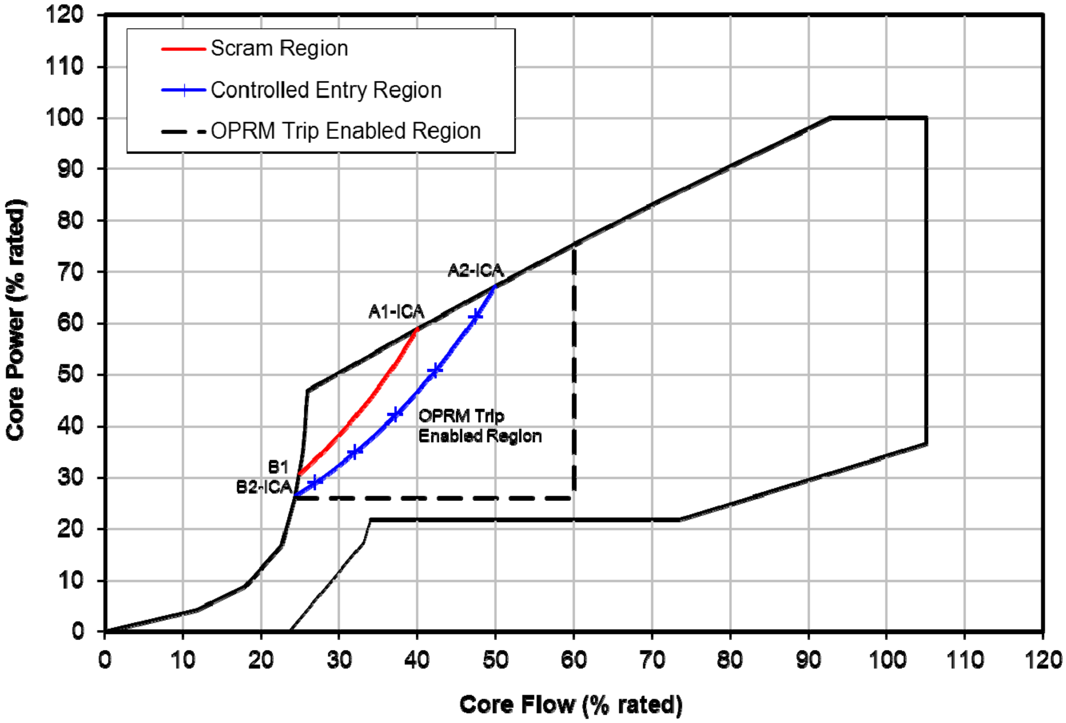

Figure 6-1 Demonstration of GGNS BSP Regions for FFWTR (320˚F Feedwater Temperature)

7.0 REFERENCES

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ATTACHMENT 4

GRAND GULF NUCLEAR STATION

GNRO-2012/00125

MARKED-UP OPERATING LICENSE PAGE

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(32) ~~Partial Feedwater Heating~~ ~~(Section 15.1, SER, SSER #2)~~

~~Operation of the plant in the partial feedwater heating mode for the purpose of extending the normal fuel cycle shall be prohibited until analyses which justify that operation are provided to and approved by the NRC staff.~~

(33) NUREG-0737 Conditions (Section 22.2)

The following conditions shall be completed to the satisfaction of the NRC. These conditions reference the appropriate items in Section 22.2, "TMI Action Plan Requirements for Applicants for Operating Licenses", in the Safety Evaluation Report and Supplements 1, 2, 3, 4, and 5 to NUREG-0831.

(a) Control Room Design Review (I.D.1, SER; Appendix E, SSER #2, SSER #4, SSER #5)

Prior to startup following the first refueling outage, SERI shall demonstrate the ability to maintain an "effective temperature" condition of 85°F or less in the remote shutdown panel (RSP) room for at least 8 hours with an ambient outdoor temperature of at least 95°F.

(b) Training During Low-Power Testing (I.G.1, SER)

Prior to restart following the first refueling outage, MP&L shall complete the additional training and testing related to TMI Action Plan I.G.1 as described in Section 2.3 of the MP&L submittal dated April 3, 1986.

(c) Deleted

(d) Hydrogen Control (Section II.B.7, SER, SSER #2, SSER #3, SSER #4, SSER #5)

(1) During the first cycle of operation, MP&L shall maintain a suitable program of analysis and testing of the installed hydrogen ignition system. EOI shall submit to the NRC quarterly reports on the status of their research programs.