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

August 31, 2015

Vice President, Operations
Entergy Operations, Inc.
Grand Gulf Nuclear Station
P.O. Box 756
Port Gibson, MS 39150

SUBJECT: GRAND GULF NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSIS PLUS (TAC NO. MF2798)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 205 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1 (GGNS). This amendment consists of changes to the facility operating license and the technical specifications (TSs) in response to your application dated September 25, 2013, as supplemented by letters dated December 30, 2013; March 10, April 11, July 31, August 14, August 26, September 4, September 10, October 2, October 20, November 20, November 21 (two letters), and December 15, 2014; and January 6, January 20, February 9, February 18, February 19, March 3, and August 13, 2015.

The amendment proposes a revision to the GGNS TSs to allow plant operation from the currently licensed Maximum Extended Load Line Limit Analysis (MELLLA) domain to plant operation in the expanded MELLLA Plus (MELLLA+) domain under the previously approved extended power uprate condition of 4408 megawatts thermal rated core thermal power.

The NRC has determined that the related safety evaluation (SE) contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations*, Section 2.390, "Public inspections, exemptions, requests for withholding." The proprietary information is indicated by text enclosed within double brackets. Accordingly, the NRC staff has also prepared a non-proprietary publicly available version of the SE, which is provided in Enclosure 2. The

NOTICE: Enclosure 3 to this letter contains Proprietary Information. Upon separation from Enclosure 3, this letter is DECONTROLLED.

proprietary version of the SE is provided in Enclosure 3. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

Alan Wang

Alan B. Wang, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures:
1. Amendment No. 205 to NPF-29
2. Safety Evaluation (non-proprietary version)
3. Safety Evaluation (proprietary version)

cc w/enclosures 1 and 2: Distribution via Listserv

<u>ENCLOSURE 1</u>

AMENDMENT NO. 205

TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

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

ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

ENTERGY MISSISSIPPI, INC.

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 205
License No. NPF-29

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated September 25, 2013, as supplemented by letters dated December 30, 2013; March 10, April 11, July 31, August 14, August 26, September 4, September 10, October 2, October 20, November 20, November 21 (two letters), and December 15, 2014; and January 6, January 20, February 9, February 18, February 19, March 3, and August 13, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-29 is hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No.205 are hereby incorporated into this license. Entergy Operations, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

The license is further amended by changes indicated in the attachment to this license amendment, and paragraphs 2.C.(48) and 2.C.(49) of Facility Operating License No. NPF-47 is hereby added to read as follows:

(48) Feedwater Heaters Out-of-Service (FWHOOS)

Operation with FWHOOS in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) region is prohibited.

(49) Time Critical Operator Action Commitments made as required fo the MELLLA+ LAR will be converted to a License Condition as follows:

Prior to Operation in the MELLLA+ Domain, Entergy will:

Train all active operating crews to perform the following three MELLLA+ time-critical operator actions:

1. Initiate Reactor Water Level Reduction (90 seconds following failure to scram concurrent with no reactor recirculation pumps in service and CTP> 5%).
2. Initiate Standby Liquid Control Injection (300 seconds if CTP> 5% or before Suppression Pool Temperature reached 110 degrees F).
3. Initiate Residual Heat Removal Suppression Pool Cooling (SPC) (660 seconds).

GGNS will validate that all active operating crews have met the time requirements for the three MELLLA+ time-critical operator actions during evaluated scenarios.

GGNS will report any MELLLA+ time-critical actions that are converted to "immediate actions" to the NRC Project Manager.

The following are one-time actions, which expire after the first report:

The results of the three MELLLA+ time-critical operator actions training will be reported to the NRC Project Manager within 60 days of completion of the training.

The reported results will include the full range of response times for each time-critical action and the average times for each crew.

3. This license amendment is effective as of its date of issuance and shall be implemented within 180 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Meena K. Khanna, Chief
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating
License No. NPF-29 and the
Technical Specifications

Date of Issuance: August 31, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 205

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Replace the following pages of the Facility Operating License No. NPF-29 and the Appendix A, Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License

Remove	Insert
4	4
16f	16f
----	16g

Technical Specifications

Remove	Insert
3.1-23	3.1-23
3.3-2a	3.3-2a
3.3-5b	3.3-5b
3.3-6	3.3-6
3.3-6a	3.3-6a
3.4-1	3.4-1
5.0-16	5.0-16
5.0-18	5.0-18
5.0-21a	5.0-21a

(b) SERI is required to notify the NRC in writing prior to any change in (i) the terms or conditions of any new or existing sale or lease agreements executed as part of the above authorized financial transactions, (ii) the GGNS Unit 1 operating agreement, (iii) the existing property insurance coverage for GGNS Unit 1 that would materially alter the representations and conditions set forth in the Staff's Safety Evaluation Report dated December 19, 1988 attached to Amendment No. 54. In addition, SERI is required to notify the NRC of any action by a lessor or other successor in interest to SERI that may have an effect on the operation of the facility.

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

Entergy Operations, Inc. is authorized to operate the facility at reactor core power levels not in excess of 4408 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 205 are hereby incorporated into this license. Entergy Operations, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

During Cycle 19, GGNS will conduct monitoring of the Oscillation Power Range Monitor (OPRM). During this time, the OPRM Upscale function (Function 2.f of Technical Specification Table 3.3.1.1-1) will be disabled and operated in an "indicate only" mode and technical specification requirements will not apply to this function. During such time, Backup Stability Protection measures will be implemented via GGNS procedures to provide an alternate method to detect and suppress reactor core thermal hydraulic instability oscillations. Once monitoring has been successfully completed, the OPRM Upscale function will be enabled and technical specification requirements will be applied to the function; no further operating with this function in an "indicate only" mode will be conducted.

(h) This license condition shall expire upon satisfaction of the requirements in paragraph (f) provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw or unacceptable flaw growth that is caused by fatigue.

(47) Commitments made as required by standard TSTF safety evaluation, as discussed in the notice of availability, will be maintained as described in UFSAR Section 16, Technical Specifications. This condition applies to the following TSTFs as approved.

TSTF-423

Changes to the commitments can be made in accordance with 10 CFR 50.59.

(48) Feedwater Heaters Out-of-Service (FWHOOS)

Operation with FWHOOS in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) region is prohibited.

(49) Time Critical Operator Action Commitments made as required for the MELLLA+ LAR will be converted to a License Condition as follows:

Prior to Operation in the MELLLA+ Domain, Entergy will:

Train all active operating crews to perform the following three MELLLA+ time-critical operator actions:

1. Initiate Reactor Water Level Reduction (90 seconds following failure to scram concurrent with no reactor recirculation pumps in service and CTP> 5%).
2. Initiate Standby Liquid Control Injection (300 seconds if CTP> 5% or before Suppression Pool Temperature reaches 110 degrees F).
3. Initiate Residual Heat Removal Suppression Pool Cooling (660 seconds).

GGNS will validate that all active operating crews have met the time requirements for the three MELLLA+ time-critical operator actions during evaluated scenarios.

GGNS will report any MELLLA+ time-critical actions that are converted to "immediate actions" to the NRC Project Manager.

The following are one-time actions which expire after the first report:

The results of the three MELLLA+ time-critical operator actions training will be reported to the NRC Project Manager within 60 days of completion of the training.

The reported results will include the full range of response times for each time-critical action and the average times for each crew.

Any MELLLA+ time-critical operator training failures during evaluated scenarios will be reported to the NRC within 60 days of any failures with a plan for resolution.

D. The facility required exemptions from certain requirements of Appendices A and J to 10 CFR Part 50 and from certain requirements of 10 CFR Part 100. These include: (a) exemption from General Design Criterion 17 of Appendix A until startup following the first refueling outage, for (1) the emergency override of the test mode for the Division 3 diesel engine, (2) the second level undervoltage protection for the Division 3 diesel engine, and (3) the generator ground over current trip function for the Division 1 and 2 diesel generators (Section 8.3.1 of SSER #7) and (b) exemption from the requirements of Paragraph III.D.2(b)(ii) of Appendix J for the containment airlock testing following normal door opening when containment integrity is not required (Section 6.2.6 of SSER #7). These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. In addition, by exemption dated December 20, 1986, the Commission exempted licensees from 10 CFR 100.11(a)(1), insofar as it incorporates the definition of exclusion area in 10 CFR 100.3(a), until April 30, 1987 regarding demonstration of authority to control all activities within the exclusion area (safety evaluation accompanying Amendment No. 27 to License (NPF-29). This exemption is authorized by law, and will not present an undue risk to the public health and safety, and is consistent with the common defense and security. In addition, special circumstances have been found justifying the exemption. Therefore, these exemptions are hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act and the rules and regulations of the Commission.

E. The licensee shall fully implement and maintain in effect all provision of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Physical Security, Safeguards Contingency and Training and Qualification Plan," and were submitted to the NRC on May 18, 2006.

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The licensee's CSP was approved by License Amendment No. 186 as supplemented by a change approved by License Amendment No. 192 and 200.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position, or can be aligned to the correct position.	31 days
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1340 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.9	Determine Boron-10 enrichment in atom percent (E).	Once within 24 hours after boron is added to the solution.
SR 3.1.7.10	Verify piping between the storage tank and the pump suction is not blocked.	Once within 24 hours after solution temperature is restored to ≥ 45°F

Next page is 3.1-26.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
J. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	J.1	Initiate action to implement the Manual BSP Regions defined in the COLR.	Immediately
	<u>AND</u>		
	J.2	Implement the Automated BSP Scram Region using the modified APRM Flow Biased Simulated Thermal Power - High trip function setpoints defined in the COLR.	12 hours
	<u>AND</u>		
	J.3	Initiate action to submit an OPRM report in accordance with Specification 5.6.7.	Immediately
K. Required Action and associated Completion Time of Condition J not met.	K.1	Initiate action to implement the Manual BSP Regions defined in the COLR.	Immediately
	<u>AND</u>		
	K.2	Reduce operation to below the BSP Boundary defined in the COLR.	12 hours
	<u>AND</u>		
	K.3	------- NOTE -------- LCO 3.0.4 is not applicable. --------------------- Restore required channels to OPERABLE.	120 days
L. Required Action and associated Completion Time of Condition K not met.	L.1	Reduce THERMAL POWER to < 16.8% RTP.	4 hours

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.20	------------------NOTE------------------ 1. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 2. For Functions 2.a, 2.b, and 2.c, the APRM/OPRM channels and the 2-Out-Of-4 Voter channels are included in the CHANNEL FUNCTIONAL TEST. 3. For Functions 2.d and 2.f, the APRM/OPRM channels and the 2-Out-Of-4 Voter channels plus the flow input function, excluding the flow transmitters, are included in the CHANNEL FUNCTIONAL TEST. -- Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.1.1.21	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.22	-----------------NOTE-------------------- For Function 2.e, "n" equals 8 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. Testing APRM and OPRM outputs shall alternate. -- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS
	DELETED	DELETED

Table 3.3.1.1-1 (page 1 of 4)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	H	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.12 SR 3.3.1.1.13	≤ 122/125 divisions of full scale
	5(a)	3	I	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.12 SR 3.3.1.1.13	≤ 122/125 divisions of full scale
b. Inop	2	3	H	SR 3.3.1.1.3 SR 3.3.1.1.13	NA
	5(a)	3	I	SR 3.3.1.1.4 SR 3.3.1.1.13	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, Setdown	2	3(c)	H	SR 3.3.1.1.7 SR 3.3.1.1.10(d)(e) SR 3.3.1.1.19 SR 3.3.1.1.20	≤ 20% RTP
b. Fixed Neutron Flux - High	1	3(c)	G	SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.10(d)(e) SR 3.3.1.1.19 SR 3.3.1.1.20	≤ 119.3% RTP
c. Inop	1,2	3(c)	H	SR 3.3.1.1.20	NA
d. Flow Biased Simulated Thermal Power - High	1	3(c)	G	SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.10(d)(e) SR 3.3.1.1.17 SR 3.3.1.1.19 SR 3.3.1.1.20	(b)(g)
e. 2-Out-Of-4 Voter	1,2	2	H	SR 3.3.1.1.19 SR 3.3.1.1.20 SR 3.3.1.1.21 SR 3.3.1.1.22	NA
f. OPRM Upscale	≥ 16.8% RTP	3(c)	J	SR 3.3.1.1.7 SR 3.3.1.1.10(d)(e) SR 3.3.1.1.19 SR 3.3.1.1.20	(f)

(continued)

Table 3.3.1.1-1 (page 2 of 4)
Reactor Protection System Instrumentation

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Two-Loop Operation: 0.64W + 61.8% RTP and ≤ 113% RTP
Single-Loop Operation: 0.58W + 37.4% RTP

(c) Each channel provides inputs to both trip systems.

(d) If the as-found channel setpoint is outside its pre-defined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(e) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance. The NTSP and the methodologies used to determine the as-found and as-left tolerances are specified in the Technical Requirements Manual.

(f) The setpoint for the OPRM Upscale Confirmation Density Algorithm (CDA) is specified in the COLR.

(g) With the OPRM Upscale trip function (Function 2.f) inoperable, reset the APRM Flow Biased Simulated Thermal Power - High trip function (Function 2.d)setpoints to the values defined by the COLR to implement the Automated BSP Scram Region in accordance with Action J of this specification.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation.

OR

One recirculation loop shall be in operation provided the plant is not operating in the MELLLA+ domain defined in the COLR and provided the required limits are modified for single loop operation as specified in the COLR.

------------------------NOTE-------------------------

Required limit modifications for single recirculation loop operation may be delayed for up to 12 hours after transition from two recirculation loop operation to single recirculation loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Recirculation loop jet pump flow mismatch not within limits	A.1 Shutdown one Recirculation loop	2 hours

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that do not meet the criteria of either Specification 5.5.11.b.1 or Specification 5.5.11.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.12 10 CFR 50, Appendix J, Testing Program

This program establishes the leakage rate testing program of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be implemented in accordance with the Safety Evaluation issued by the Office of Nuclear Reactor Regulation dated April 26, 1995 (GNRI-95/00087) as modified by the Safety Evaluation issued for Amendment No. 135 to the Operating License, except that the next Type A test performed after the November 24, 1993 Type A test shall be performed no later than November 23, 2008. Consistent with standard scheduling practices for Technical Specifications required surveillances, intervals for the recommended surveillance frequency for Type A, B and C testing may be extended by up to 25 percent of the test interval, not to exceed 15 months. The calculated peak containment internal pressure for the design basis loss of coolant accident, Pa, is 12.1 psig.

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and process control program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Deleted

5.6.5 Core Operating Limits Report (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1) LCO 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR),

2) LCO 3.2.2, Minimum Critical Power Ratio (MCPR),

3) LCO 3.2.3, Linear Heat Generation Rate (LHGR),

4) Deleted

5) LCO 3.3.1.1, RPS Instrumentation, Table 3.3.1.1-1 APRM Function 2.f

6) The Manual Backup Stability Protection (BSP) Scram Region (Region 1), the Manual BSP Controlled Entry Region (Region II), the modified APRM Flow Biased Simulated Thermal Power - High trip function (Function 2.d) setpoints used in the OPRM Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1.

(continued)

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

i) Limiting Conditions for Operations Section 3.4.11, "RCS Pressure and Temperature (P/T) Limits"

ii) Surveillance Requirements Section 3.4.11, "RCS Pressure and Temperature (P/T) Limits"

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

i) NEDC-33178P-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Temperature Curves" Revision 1, June 2009

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Oscillation Power Range Monitor (OPRM) Report

When an OPRM report is required by CONDITION J of LCO 3.3.1.1, "RPS Instrumentation," it shall be submitted within the following 90 days. The report shall outline the preplanned means to provide backup stability protection, the cause of the inoperability, and the plans and schedule for restoring the required instrumentation channels to OPERABLE status.

<u>ENCLOSURE 2</u>

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO. 205 TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

Proprietary information pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* has been redacted from this document.

Redacted information is identified by blank space enclosed within double brackets.

ENCLOSURE 2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR

REGULATION RELATED TO AMENDMENT NO. 205

TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

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GRAND GULF NUCLEAR STATION, UNIT 1

SAFETY EVALUATION FOR MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSIS PLUS

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 205 TO

FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC., ET AL.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By application dated September 25, 2013 (Reference 1), as supplemented by letters dated December 30, 2013 (Reference 2); March 10, April 11, July 31, August 14, August 26, September 4, September 10, October 2, (Reference 3 through 10); October 20 (e-mail Reference 11), November 20, November 21 (two letters), and December 15, 2014 (Reference 12-15) and January 6, January 20, February 9, February 18, February 19, March 3, 2015 (References 16 through 21), and August 13, 2015 (Reference 67), Entergy Operations, Inc. (Entergy, the licensee), requested changes to the technical specifications (TSs) for Grand Gulf Nuclear Station, Unit 1 (GGNS). The December 30, 2013, letter was in response to a letter dated December 19, 2013, "Supplemental Information Needed for Acceptance of Licensing Action, Request to Allow Operation in Expanded Maximum Extended Load Line Limit Analysis Plus Domain," which provided additional information to complete the acceptance review.

The license amendment request (LAR) (Reference 1), proposes a revision to the GGNS TSs to allow plant operation from the currently licensed Maximum Extended Load Line Limit Analysis (MELLLA) domain to plant operation in the expanded MELLLA Plus (MELLLA+) domain under the previously approved extended power uprate (EPU) conditions of 4408 megawatts thermal (MWt) rated core thermal power.

The supplemental letters dated July 31, August 14, August 26, September 4, September 10, October 2, November 20, November 21 (two letters), and December 15, 2014; and January 6, January 20, February 9, February 18, February 19, March 3, and August 13, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 2, 2014 (79 FR 71453). The NRC staff's notice considered the September 25, 2013 application and supplemental letters dated December 30, 2013, March 10, 2014, and April 11, 2014.

1.1 Background

GGNS is a boiling-water reactor (BWR) plant of the BWR/6 design with a Mark-III containment. The NRC licensed GGNS on November 1, 1984, under NPF-29 (Reference 22), for full-power operation at the original licensed thermal power (OLTP) of 3833 MWt, and it entered commercial operation on July 1, 1985. In License Amendment No. 156 dated October 10, 2002 (Reference 23), the GGNS licensed thermal power limit was increased by approximately 1.7 percent from 3833 MWt to 3898 MWt (i.e., the current power level). The 1.7 percent power change was based on the installation of the Caldon Leading Edge Flow Meter ultrasonic flow measurement system and its ability to achieve increased accuracy in measuring feedwater (FW) flow. An EPU, which increased the power level by 15 percent, was approved by License Amendment No. 191 dated July 18, 2012 (Reference 24), for GGNS that increased the power level to 4408 MWt.

GGNS is located in Claiborne County, Mississippi, on the east bank of the Mississippi River at River Mile 406, approximately 25 miles south of Vicksburg, Mississippi, and 37 miles north-northeast of Natchez, Mississippi. Port Gibson, located approximately 6 miles to the southeast, is the closest town to GGNS, with a 2013 Census population of 1,466.

The construction permit for GGNS was issued by the Atomic Energy Commission on September 4, 1974 (Reference 25). The plant was designed and constructed based on Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "General Design Criteria [GDC] for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971 (Reference 26) (hereinafter referred to as "final GDC"). The 64 GDC establish minimum requirements for the principal design criteria for water-cooled nuclear power plants, including GGNS.

As discussed in the GGNS Updated Final Safety Analysis Report (UFSAR) (Reference 27), Section 3.1, "Conformance with the NRC General Design Criteria," for each of the 64 GDC, a specific assessment of the plant design has been made. In addition, a list of the GGNS UFSAR sections with further information pertinent to each criterion is also provided.

1.2 Licensee's Approach

In its September 25, 2013, application (Reference 1) to operate GGNS in the MELLLA+ expanded operating domain, Entergy submitted Licensing Topical Report (LTR) NEDC-33612P, Revision 0, "Safety Analysis Report (MELLLA+ Safety Analysis Report (SAR) for GGNS Maximum Extended Load Line Limit Analysis Plus," (Reference 28). MELLLA+ is an extension of the reactor operating domain. Under MELLLA+, the operating power is maintained constant, but the recirculation core flow is allowed to operate within a wider window than under MELLLA. For GGNS, the MELLLA+ flow window is between 80 percent and 105 percent flow. This operating flexibility reduces the need for frequent control rod motion. A secondary effect of MELLLA+ is increased fuel utilization caused by increased Plutonium (Pu) production with increased void fraction levels, which hardens the neutron flux spectrum.

The current operating core in GGNS contains only General Electric Hitachi (GEH) Global Nuclear Fuel (GNF) fuel product line GNF2. The NRC staff approved the applicability of GEH methods to expanded operating domains-supplement for GNF2 Fuel, on December 28, 2010 (Reference 29). The SAR calculations are based on a full equilibrium core of GNF2 fuel.

In LTR NEDC-33612P, Revision 0 (Reference 28), the licensee documents the results of all significant safety evaluations (SEs) performed to justify the expansion of the core flow operating domain for GGNS to the MELLLA+. These analyses support the operation of GGNS at the post-EPU current licensed thermal power (CLTP) of 4408 MWt with core flow as low as 80 percent of rated flow. All topics in the SAR are dispositioned as either "Generic (see Section 3.2 of the SE)" or "Plant-specific (see Section 3.3 of the SE)" as outlined in NEDC-33006P-A, the MELLLA+ Safety Evaluation Report (SER).

These licensees' analyses are based on the approved methodologies identified in the following NRC staff-approved LTRs:

The MELLLA+ Safety Evaluation Report, NEDC-33006P-A, Revision 3, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus" (Reference 30).

This LTR (MELLLA+ SER) evaluates the impact of operation in the expanded operating domain on BWRs regarding (1) safety systems and components capability and performance, and (2) response to the design bases and special events that demonstrate plants can meet the regulatory and safety requirements. The MELLLA+ SER dispositions the principal review topics generically or proposes that plant-specific analyses will be provided in the MELLLA+ applications to quantify the impact.

The Methods to Expanded Operating Domains Safety Evaluation Report, LTR NEDC-33173P-A, "Applicability of GE [General Electric] Methods to Expanded Operating Domains," Revision 4 (Reference 31).

This LTR extends the use of GEH's analytical methods and codes to MELLLA+ domain. Plant-specific MELLLA+ applications must demonstrate compliance with the limitations in the NRC staff's SE approving NEDC-33173P-A, or any supplements or revisions.

The Detect and Suppress Solution-Confirmation Density Safety Evaluation Report, NEDC-33075P, "Detect and Suppress Solution-Confirmation Density [DSS-CD] Licensing Topical Report," Revision 7 and NEDC-33075P-A, "Detect and Suppress Solution-Confirmation Density [DSS-CD] Licensing Topical Report," Revision 8 (Reference 32).

This LTR presents stability detect and suppress methodology for application to MELLLA+ operation. The NRC staff reviewed and approved the stability methodology presented in this LTR for application to MELLLA+ operation. Specifically, the NEDC-33075P-A stability detect and suppress methodology ensures that the stability response for operation at the higher MELLLA+ rod line can be detected and suppressed such that Appendix A to Part 50 of 10 CFR, GDC 12, "Suppression of Reactor Power Oscillations," requirements can be met.

DSS-CD TRACG Application SER, NEDE-33147P-A, "DSS-CD TRACG Application," Revision 4 (Reference 33).

GEH used TRACG calculations to demonstrate that the DSS-CD stability solution can effectively detect and suppress instability events and meet the associated regulatory requirements. The NRC staff reviewed and accepted TRACG for this specific application.

All limitations from the LTRs listed above have been addressed in this SE and are discussed in Section 3.8 of this SE.

1.3 Method of NRC Review

The NRC staff's review is based on the submitted LAR, which includes the MELLLA+ SAR, the information obtained during a number of meetings and conference calls with the licensee, and formal requests for additional information (RAIs). To evaluate the impact of operation in the expanded operating domain, the staff performed this review using relevant sections of the review guidance in Review Standard 001 (RS-001), Revision 0, "Review Standard for Extended Power Uprates" (Reference 34), relevant sections of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants" (Reference 35), and the findings of the MELLLA+ staff's SER, NEDC-33006P-A, Revision 3, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus" (Reference 30).

The NRC staff concludes from the review of this LAR that the broadening of the GGNS operating domain by lowering the flow at high power without additional limitations would reduce the safety margin, but the solutions proposed by the licensee in the SAR (Reference 28) are technically acceptable to satisfy the regulatory criteria. The following solutions are proposed by the licensee to maintain the same safety margin under operating in the MELLLA+ domain rather than under the MELLLA domain under the CLTP:

1. FW heater out-of-service (FWHOOS) operation will not be allowed in the MELLLA+ domain because analyses have not been performed to demonstrate compliance with applicable criteria under these conditions.

2. Single-loop operation (SLO) is not allowed in the MELLLA+ domain.

3. The discharge pressure for the standby liquid control system (SLCS) pump will be increased to accommodate larger transient over-pressure.

4. To provide additional protection against spurious, noise-induced scram of the DSS-CD system, the amplitude discriminator setpoint (S_{AD}) **[[**

   ]], based on the methodology described in Section 2.4 of the SAR). **[[**

]] the process described in Section 2.4 of the SAR and in Section 6 of the DSS-CD LTR (Reference 32).

5. Typically, the limiting anticipated operational occurrences (AOOs) result in larger delta-critical power ratio (ΔCPR) when initiated at nominal conditions, than when initiated at lower flows inside the MELLLA+ domain. However, the results of the licensee's analyses (summarized in Table 9-1 of the SAR (Reference 28) indicates that for the turbine trip and load rejection AOOs, the calculated ΔCPR is more limiting at lower flows in GGNS. Therefore, additional OLMCPR margin is required for GGNS to operate in the MELLLA+ domain. This additional margin is incorporated during the reload analysis process.

The GGNS MELLLA+ anticipated transient without scram (ATWS) with instability (ATWS-I) calculation satisfies the ATWS acceptance criteria (Reference 35(d), Section 15.8), in part, by taking one deviation from the standard methodology in the MELLLA+ SER, NEDC-33006P-A (Reference 30):

> Operator actions to reduce reactor vessel water have been assumed to occur within 90 seconds of the ATWS initiation. This is faster than the recommended value of 120 seconds in NEDC-33006P-A and faster than the conservative time used for past ATWS analyses of 250 seconds.

To minimize the consequences to fuel of unstable power oscillations during ATWS events with instability, the licensee has committed (References 1, 18, and 20) to train the reactor operators to initiate FW flow reduction within 90 seconds of event initiation. This action causes a prompt reactor water level decrease that tends to minimize the amplitude of the unstable power oscillations. This is also discussed in Section 3.3 (SAR Section 10.6); and Section 3.7, "Special Events," Subsection, "ATWS-I," of this SE.

The NRC staff determined that the licensee has adequately accounted for the effects of the proposed MELLLA+ operating domain extension on the nuclear design and demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the pressure vessel or impair the capability to cool the core. Details of the staff's review of the SAR are provided in Section 3.0 of this SE.

2.0 REGULATORY EVALUATION

The NRC staff's regulatory criteria in this review are based on the following sources (see Section 10 References):

> Review Standard 001 (RS-001) (Reference 34).

- Regulatory Guide (RG) 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing" (Reference 37).

- RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 38).

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 39).

- Relevant sections of the SRP (Reference 35), specifically:

(a) Chapter 3, "Design of Structures, Components, Equipment, and Systems"
  - Section 3.9.1, "Special Topics for Mechanical Component"
  - Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components"
  - Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures"
  - Section 3.9.5, "Reactor Pressure Vessel Internals"

(b) Chapter 4, "Reactor"
  - Section 4.2, "Fuel System Design"
  - Section 4.3, "Nuclear Design"
  - Section 4.4, "Thermal and Hydraulic Design"
  - Section 4.6, "Functional Design of Control Rod Drive System"

(c) Chapter 5,
  - Section 5.2.2, "Overpressure Protection"
  - Section 5.4.6, "Reactor Core Isolation Cooling (BWR)"
  - Section 5.4.7, "Residual Heat Removal (RHR) System"

(c) Chapter 6, "Engineered Safety Features"
  - Section 6.2.1.1.C, "Pressure-Suppression Type BWR Containments
  - Section 6.2.1.2, "Subcompartment Analysis"
  - Section 6.2.2, "Containment Heat Removal Systems"
  - Section 6.2.5, "Combustible Gas Control in Containment"

(d) Chapter 15, "Transient and Accident Analysis"
  - Section 15.1, "Increase in Heat Removal by the Secondary System"
  - Section 15.2, "Decrease in Heat Removal by the Secondary System"
  - Section 15.3, "Decrease in Reactor Coolant System Flow Rate"
  - Section 15.4, "Reactivity and Power Distribution Anomalies"
  - Section 15.5, "Increase in Reactor Coolant Inventory"
  - Section 15.6, "Decrease in Reactor Coolant Inventory"
  - Section 15.7, "Radioactive Release from a Subsystem or Component"

- Section 15.8, "Anticipated Transients Without Scram"
- Section 15.9, "Boiling Water Reactor Stability"

(e) Chapter 19, "Severe Accidents"
- Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," Appendix D

- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," which establishes standards for the calculation of emergency core cooling accident (ECCS) performance and acceptance criteria for that calculated performance.

- Appendix K to 10 CFR Part 50, "ECCS Evaluation Models," which establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a loss-of-coolant accident (LOCA).

- 10 CFR 50.36(c)(3), which states "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and the limiting conditions for operation will be met." There are specifications that the Commission established in its regulatory requirements related to the contents of the TSs. Specifically, 10 CFR 50.36(a)(1) states, in part, "[e]ach applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section."

- 10 CFR 50.44, "Combustible gas control for nuclear power reactors," insofar as it requires that plants be provided with the capability of controlling combustible gas concentrations in the containment atmosphere.

- 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," insofar as licensees provide the means to address an ATWS event, an AOO defined in Appendix A of Part 50, followed by the failure of the reactor trip portion of the protection system specified in GDC 20 of Appendix A to Part 50, "Protection systems function."

- 10 CFR 50.63, "Loss of all alternating current power," insofar as it requires that the plant withstand and recover from a station blackout (SBO) event of a specified duration.

- 10 CFR 50.67, "Accident source term," requirements for licensees "who seek to revise the current accident source term used in their design basis radiological analyses."

- NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3 (Reference 40).

- NUREG-1764, "Guidance for the Review of Changes to Human Actions;" Revision 1 (Reference 41).

The NRC staff's acceptance criteria are based on the following GDC in Appendix A of 10 CFR Part 50:

- GDC 1, "Quality standards and records," insofar as it requires those structures, systems and components (SSCs) important to safety shall be designed, fabricated, erected, tested, to quality standards commensurate with the importance of the safety functions to be performed.

- GDC 2, "Design bases for protection against natural phenomena," insofar as it requires SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

- GDC 4, "Environmental and dynamic effects design bases," insofar as it requires that SSCs important to safety must be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer.

- GDC 5 "Sharing of structures, systems and components," insofar as it requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions.

- GDC 10, "Reactor design," insofar as the reactor protection system (RPS) shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

- GDC 11, "Reactor inherent protection," insofar as the reactor core must designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

- GDC 12, "Suppression of reactor power oscillations," insofar as the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

- GDC 13, "Instrumentation and control," insofar as instrumentation and controls must be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

- GDC 15, "Reactor coolant system design," insofar as it requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs).

- GDC 16, "Containment design," insofar as it requires that the reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

- GDC 19, "Control room," insofar as it requires that adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

- GDC 20, "Protection system functions," insofar as it required that the protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

- GDC 23, "Protection system failure modes," insofar as it requires that the protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

- GDC 25, "Protection system requirements for reactivity control malfunctions," insofar as it requires that he protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

- GDC 26, "Reactivity control system redundancy and capability," insofar as two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

- GDC 27, "Combined reactivity control system capability," insofar as it requires that the reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

- GDC 28, "Reactivity limits," insofar as it requires that the reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

- GDC 29, "Protection against anticipated operational occurrences," insofar as it requires that the protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs.

- GDC 31, "Fracture prevention of reactor coolant boundary," insofar as it requires that the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

- GDC 33, "Reactor coolant makeup," insofar as it requires that a system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

- GDC 34, "Residual heat removal," insofar as it requires that a system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

- GDC 35, "Emergency core cooling," insofar as it requires that a system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

- GDC 38, "Containment heat removal," insofar as it requires that a system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

- GDC 40, "Testing of containment heat removal system," insofar as it requires that the containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

- GDC 41, "Containment atmosphere cleanup," insofar as it requires that a systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

- GDC 42, "Inspection of containment atmosphere cleanup systems," insofar as it requires that the containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

- GDC 50, "Containment design basis," insofar as it requires the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by 10 CFR 50.44, energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

- GDC 54, "Piping systems penetrating containment," insofar as it requires that piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

3.0 TECHNICAL EVALUATION

3.1. Overview of NEDC-33612P

The GGNS MELLLA+ SAR, NEDC-33612P, Revision 0 (Reference 28), contains information divided into the following 11 sections:

SAR Section 1.0 – Introduction
SAR Section 2.0 – Reactor Core and Fuel Performance
SAR Section 3.0 – Reactor Coolant and Connected Systems
SAR Section 4.0 – Engineered Safety Features
SAR Section 5.0 – Instrumentation and Control
SAR Section 6.0 – Electrical Power and Auxiliary Systems
SAR Section 7.0 – Power Conversion Systems
SAR Section 8.0 – Radwaste Systems and Radiation Sources
SAR Section 9.0 – Reactor Safety Performance Evaluations
SAR Section 10.0 – Other Evaluations
SAR Section 11.0 – Licensing Evaluations

The SAR also includes three appendices that evaluate the disposition of limitations of applicable SERs. A complete listing of the required limitations and conditions is presented in Appendices A, B, and C of the SAR. These appendices address the limitations from the MELLLA+ SER (Reference 30), the methods SER (Reference 31), and the DSS-CD SER (Reference 32). Note that in prior MELLLA+ applications, a fourth appendix was included to account for one limitation of the TRACG application for DSS-CD (Reference 33). However, the new Revision 7 of the DSS-CD-SER incorporates the TRACG application and the old limitation no longer applies. Therefore, a fourth appendix was not necessary for GGNS.

3.1.1. SAR Section 1.0, "Introduction"

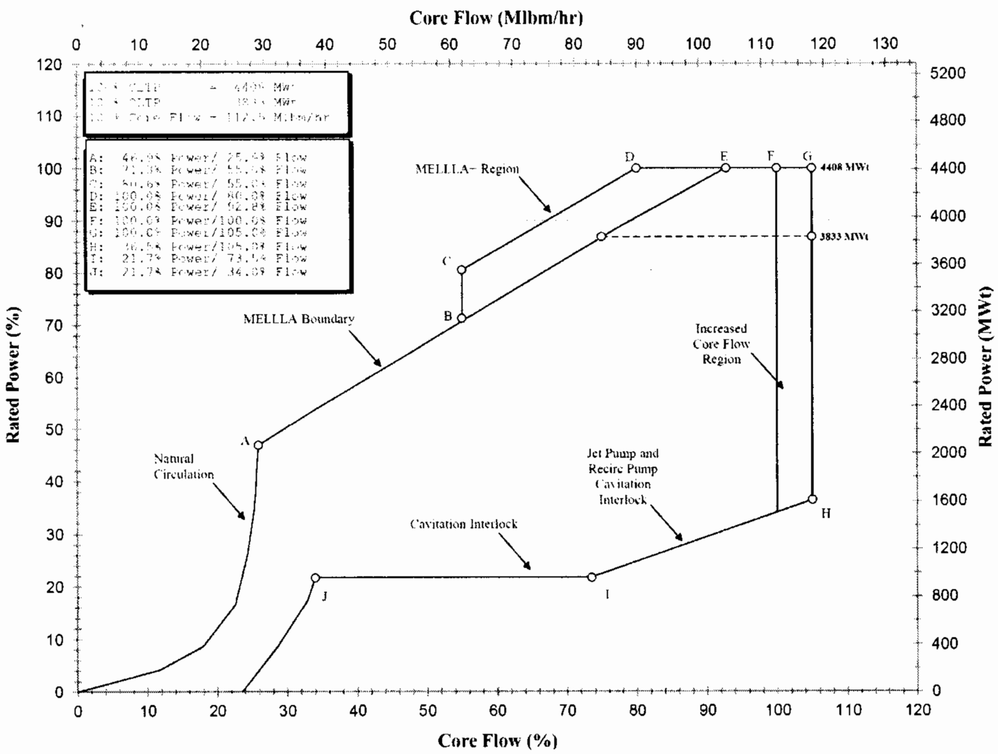
Section 1.0, "Introduction," of the SAR describes the report approach, as well as the differences between generic and plant-specific assessments. Generic assessments are those topics that can be disposed of by either (1) a reference bounding calculation, (2) demonstration of a negligible impact from MELLLA+ operation, (3) or deferring to the plant-specific analyses during the reload process. Plant-specific evaluations are provided for those items where a generic assessment is not applicable. However, for some systems and components, a plant-specific evaluation was needed to disposition that the system or component would not be affected by operating in the MELLLA+ domain. These topics are discussed in Section 3.3 of this SE.

The licensee committed to supplement the SAR with the fuel and cycle dependent analyses, including the plant-specific thermal limits assessment. The NRC staff reviewed the Supplemental Reload Licensing Report (SRLR) (Reference 42) for the initial MELLLA+ implementation (Cycle 20) and confirmed that the analyses documented in the SRLR support the conclusions in the SE.

Table 1-1 of the GGNS's SAR lists all of the computer codes used in the MELLLA+ SAR evaluations. Figure 1-1 of the SAR (reproduced here as Figure 1 below) defines the MELLLA+ operating domain.

Figure 1. MELLLA+ Operating Domain for GGNS

(The upper boundary of the domain is defined by the following relation between the power (P), and the percent core flow (W_T), where the GGNS plant-specific lower limit for W_T is 54%.)
(P = 100 + 0.7749 (W_T – 80%)



SAR Section 1.2.4, "Operational Enhancements," describes the allowed operational enhancements, which are covered by the approved MELLLA+ SER.

The following enhancements are not allowed in the MELLLA+ domain:

1. FWHOOS
2. SLO

SAR Section 2.0, "Reactor Core and Fuel Performance," states that because GGNS will use only GNF2 fuel during MELLLA+ operation, the following limitations and conditions from the methods LTR SER are not applicable to GGNS:

(a) Application of 10 Weight Percent gD [Gadolinium]: Limitation and Condition 9.13
(b) Mixed Core Method 1: Limitation and Condition 9.21
(c) Mixed Core Method 2: Limitation and Condition 9.22

However, because GGNS introduces GNF2 fuel for the first time in a MELLLA+ application, the following limitations from the MELLLA+ SER and DSS-CD SER are applicable:

- *MELLLA+ SER Limitations and Conditions:*

  (a) Concurrent Changes: Limitation and Conditions 12.3.d, 12.3.e, and 12.3.f
  (b) Appendix A RAI 14-9: Limitation and Condition 12.23.6
  (c) Appendix A RAI 14-10: Limitation and Condition 12.23.7

- *DSS-CD SER Limitations and Conditions:*

  Limitation and Condition 5.4. Attachment C to the GGNS SAR addresses this limitation and Condition and confirms the DSS-CD trip function is applicable to GGNS.

SAR Section 2 addresses additional limitations and conditions related to the reactor core and fuel performance, including:

- TGBLA/PANAC Version. The most recent version (TGBLA06/PANAC11) at the time of analyses was used (Methods SER Limitation and Condition 9.1).

- MELLLA+ LTR SER Limitation and Condition 12.24.1. The TRACG supporting analyses used the detailed calculation of bundle flow as required by this condition.

SAR Section 2 also provides, as a function of the cycle exposure, a comparison of GGNS fuel performance versus other plants and cycles in Figures 2-1 through 2-6 for peak bundle power, peak bundle flow, peak linear heat generation rates (LHGR), and exit void fraction for the peak bundle, maximum void, and core average.

SAR Section 2 provides the power distributions, LHGR, and critical power ratios (CPRs) at three points during the fuel cycle.

3.2. Generic MELLLA+ Dispositions

As discussed above, the following topics were evaluated generically in the approved methodology, as described in the GE MELLLA+ SER (Reference 30). As required by the MELLLA+ SER, Section 2.0 of the SAR evaluates the topics and confirms that these evaluations are applicable to GGNS. For the topics that were dispositioned generically, the GGNS SAR stated that operation in the MELLLA+ domain was justified based on the following:

- Provided or referenced a bounding analysis for the limiting condition;
- Demonstrated that there is a negligible effect;
- The system or component is unaffected by the MELLLA+ power/flow map operating domain expansion; or
- The sensitivity to MELLLA+ is small enough that the required plant cycle-specific reload analysis process is sufficient and appropriate for establishing the MELLLA+ licensing basis.

The NRC staff has summarized the licensee's discussions regarding the disposition of the SAR topics in Sections 3.2.1 to 3.2.9 of this SE. The numbering of the topics in each subsection of this SE is consistent with the GGNS SAR numbering. Section 3.2.10 of this SE provides the staff's conclusion of the licensee's generic MELLLA+ dispositions.

As indicated in Section 3.1.1 of this SE, the plant-specific evaluations are addressed in Section 3.3 of this SE.

3.2.1. SAR Section 2.0, "Reactor Core and Fuel Performance"

The licensee concludes that no plant-specific evaluations are required for Section 2, "Reactor Core and Fuel Performance," areas, as GGNS meets all conditions of the MELLLA+ SER for generic disposition. Even though the licensee concludes that the topics in this section meet the generic disposition of the MELLLA+ SER, the NRC staff performed a review to confirm these conclusions. The NRC staff used RS-001 (Reference 34) as a reference in conducting the MELLLA+ review. The following subsections of the SAR Section 2, are addressed in this SE as follows:

SAR Section 2.1, "Fuel and Operation" – SE Sections 3.4.1 and 3.4.2
SAR Section 2.2, "Thermal Limits Assessment" - SE Sections 3.4.2 and 3.4.3
SAR Section 2.3, "Reactivity Characteristics" – SE Section 3.4.2
SAR Section 2.4, "Stability" – SE Section 3.4.3, "Subsection Stability"
SAR Section 2.5, "Reactivity Control" – SE Section 3.4.2
SAR Section 2.6, "Additional Limitations and Conditions Related to Reactor Core and Fuel Performance" – SE Section 3.3.7

The NRC staff reviewed the impact on the fuel system of the proposed MELLLA+ operating domain extension based on the licensee-provided analyses for normal operation, AOOs, and special events. The complete NRC staff evaluation of these results is documented in Sections 3.6 and 3.7 of this SE. As seen in these evaluations, operation at the lower MELLLA+ flows has an impact on transient response, and the effect on fuel becomes slightly more severe for some events. To mitigate these events, the licensee proposes to use more restrictive setpoints consistent with the AOO ΔCPR methodology, so that the final safety limit minimum critical power ratio (SLMCPR) limit is maintained. These SLMCPR setpoint changes supporting MELLLA+ operation were approved by License Amendment No. 203, dated August 18, 2015 (ADAMS Accession No. ML15203A071). The licensee analyses demonstrate that, with the proposed GGNS MELLLA+ setpoints, fuel damage is not expected for any AOO or analyzed special events, and core coolability is always maintained. Thus, the NRC staff concludes that the impact on fuel, while operating with the more restrictive setpoints at the lower MELLLA+ flows, is minimal.

3.2.2. SAR Section 3.0, "Reactor Coolant and Connected Systems"

SAR Section 3.1.1, "Flow-Induced Vibration"

Implementation of MELLLA+ does not increase main steam line (MSL) flow; therefore, there is no effect on the flow-induced vibration (FIV) of the piping and safety relief valves (SRVs).

SAR Section 3.2, "Reactor Vessel"; SAR Section 3.2.2, "Reactor Vessel Structural Evaluation"

Implementation of MELLLA+ does not change the reactor operating pressure, maximum FW flow rates, or steam flow rates; therefore, there is no change to the stress or fatigue for reactor vessel components.

SAR Section 3.3, "Reactor Internals"; SAR Section 3.3.1, "Reactor Internal Pressure Differences"; SAR Section 3.3.1.1, "Fuel Assembly and Control Rod Guide Tube Lift Forces"

The only variable change affecting reactor internal pressure differences (RIPDs) is core flow (CF). As maximum CF is reduced in the MELLLA+ operating domain, the RIPDs for normal, upset emergency, and faulted conditions, are bounded by the current licensed operating domain (CLOD).

SAR Section 3.3.1.2, "Reactor Internals Pressure Differences for Normal, Upset, Emergency and Faulted Conditions"

The only variable change affecting reactor internal pressure differences (RIPDs) is core flow (CF). As maximum CF is reduced in the MELLLA+ operating domain, the RIPDs for normal, upset emergency, and faulted conditions, are bounded by the CLOD.

SAR Section 3.4, "Flow-Induced Vibration"; SAR Section 3.4.1, "FIV Influence on Piping"

Implementation of MELLLA+ does not increase the flow rates in the recirculation, main stream (MS), or FW lines; therefore, there is no increase in FIV in these piping systems and is bounded by the current licensing basis.

SAR Section 3.4.2, "FIV Influence on Reactor Internals"

Implementation of MELLLA+ [[

]]. The NRC staff's review of the steam dryer is addressed in Section 3.3, and SAR Section 3.3.2, of this SE.

SAR Section 3.5, "Piping Evaluation"; SAR Section 3.5.1.1, "Main Steam and Feedwater Piping Inside Containment"

System pressures, temperatures and flows in the MELLLA+ operating domain are within the range of the current range of rated operating parameters for MS and FW piping inside containment. As such, these systems inside containment are unaffected by operation in the MELLLA+ domain.

SAR Section 3.5.1.2, "Reactor Recirculation and Control Rod Drive Systems"

Implementation of MELLLA+ does not change the maximum operating system temperatures, pressures, and flows for the recirculation piping system and residual heat removal (RHR) piping system. As such, [[

]]
system as result of operation in the MELLLA+ domain. This CRD system is further discussed in Section 3.4.1 of this SE.

SAR Section 3.5.1.3, "Other RCPB Piping Systems"; SAR Section 3.5.1.3.1, "Other RCPB Piping Systems – HPCS [High Pressure Core Spray], LPCS [Low Pressure Core Spray], RHR/LPCI [Low Pressure Coolant Injection], and SLCS"; SAR Section 3.5.1.1.3.2, "Other RCPB Piping Systems – RPV Head Vent Line and SRV Discharge Lines"; SAR Section 3.5.1.3.3, "Other RCPB Piping – RWCU [Reactor Water Cleanup]"; SAR Section 3.5.1.3.4, "Other RCPB Piping Systems – Safety Related Thermowells"

For implementation of the above Sections, MELLLA+ does not affect the maximum operating temperature, pressure, or flow rate of any of the following systems: HPCS, LPCS, RHR/LPCI, SLCS, and other RCPB piping systems. In addition, as these systems are isolated by containment isolation valves during normal operation, there is zero flow in these systems during normal plant operation. As such, the system temperatures, flows, and pressures at MELLLA+ conditions are bounded by the CLTP temperatures, flows, and pressures. They are within the design values used in the design of the piping and supports chosen for the worst case conditions. Therefore, they are unaffected by the MELLLA+ operating domain expansion, and their susceptibility to erosion/corrosion does not increase.

SAR Section 3.5.1.4, "Other Than Category "A" RCPB Material"

Category "A" materials exist in the RCPB piping." Category "A" is a resistant material to IGSCC for BWR piping weldments in accordance with NUREG-0313 (Reference 43). "Other than Category A" is assumed to mean non-resistant or cracked materials for intergranular stress corrosion cracking (IGSCC) BWR weldments in accordance with NUREG-0313 (IGSCC Categories B through G).

Entergy has implemented at GGNS an inservice inspection program for RCPB piping that is in accordance with the American Society of Mechanical Engineers (ASME), Section XI, (Reference 44) coupled with the augmented program for reactor coolant piping based on Generic Letter (GL) 88-01 (Reference 45), NUREG-0313 (Reference 43), and the Boiling Water Reactor Vessel Internals Project (BWRVIP)-75-A (Reference 46). The augmented inspection program is designed to detect potential degradation from IGSCC. Entergy has concluded that stress improvement processes and the original construction processes used for IGSCC resistance are not affected by operating in the MELLLA+ domain. Also, GGNS has implemented hydrogen water chemistry, which reduces the potential for IGSCC of RCPB components. Therefore, the augmented inspection program at GGNS is adequate to address concerns related to "other than Category A" materials in the RCPB.

SAR Section 3.5.2, "Balance-of-Plant Piping" (BOP); SAR Section 3.5.2.1, "Main Steam and Feedwater (Outside Containment)"

As stated in the GGNS SAR, implementation of MELLLA+ "does not increase the maximum operating temperature, pressure, flow rate, or mechanical loads for the MS and FW piping outside containment. MS and FW system temperatures, flows, and pressures at MELLLA+ conditions are bounded by the CLTP temperatures, flows, and pressures, and as such are within the design values used in the design of the piping and supports chosen for worst case conditions. As such, the GGNS MS and FW piping outside containment is unaffected by the operation in the MELLLA+ domain.

SAR Section 3.5.2.2, "Other BOP Piping Systems"; SAR Section 3.5.2.2.1, "Other BOP Piping Systems – RCIC, HPCS, LPCS, and RHR"

Implementing the MELLLA+ does not change the maximum operating temperature, pressure, or flow rate, nor increase mechanical loads for any of the following systems: reactor core isolation cooling (RCIC), HPCS, LPCS, and RHR. RCIC, HPCS, LPCS, and RHR system temperatures, flows, and pressures at MELLLA+ conditions are bounded by the CLTP temperatures, flows, and pressures. As such are within the design values used in the design of the piping and supports chosen for the worst case conditions. The above components are, therefore, unaffected by operation in the MELLLA+ domain.

SAR Section 3.5.2.2.2, "Other BOP Piping Systems - Offgas System, and Neutron Monitoring System"

Implementation of MELLLA+ does not change the GGNS reactor operating pressure or power level; therefore, these systems are unaffected by operation in the MELLLA+ domain.

SAR Section 3.6, "Reactor Recirculation System"; SAR Section 3.6.1, "System Evaluation"

The GGNS RRS operating conditions in the MELLLA+ operating domain are within the operating conditions in the CLOD. For GGNS, there are no increases in the RRS temperature or flow rates as a result of MELLLA+ operating domain expansion as compared to the CLOD. For BWR/6 plants, with a constant dome pressure, the RRS system pressure will increase at MELLLA+ operating conditions due to flow control valve closure. However, this pressure increase is within the system design parameters. RRS system temperature for the CLOD is 530.1 degrees Fahrenheit (°F) and in the MELLLA+ operating domain, it is 526.6 °F. For GGNS, SLO is not allowed in the MELLLA+ operating domain. **[[**

]].

SAR Section 3.6.2, "Net Positive Suction Head"

[[

]].

SAR Section 3.6.3, "Single Loop Operation"

SLO is not allowed in the MELLLA+ operating domain. SLO is limited to the MELLLA region of the power/flow map as shown in Figure 4-3 of the Core Operating Limits Report (COLR) as directed per TS 3.4.1. Therefore, SLO is not allowed in the MELLLA+ operating range and is not affected by the MELLLA+ domain expansion in the MELLLA+ operating domain.

SAR Section 3.7, "Main Steam Line Flow Restrictors"

There is no increase in GGNS MS flow as a result of the MELLLA+ operating domain expansion. Thus, **[[**

]] as a result of operating in the MELLLA+ domain.

SAR Section 3.8, "Main Steam Isolation Valves"

There is no significant increase in GGNS MS pressure, flow, or pressure drop as a result of MELLLA+ operating domain expansion. The total MSL pressure drop at the turbine stop valves is not significantly changed for MELLLA+. The main steam isolation valve (MSIV) pressure drop is also not significantly changed. **[[**

]] as a result of operating in the MELLLA+ domain.

SAR Section 3.9, "Reactor Core Isolation Cooling"; SAR Section 3.91, "System Hardware"; SAR Section 3.92, "System Initiation"; and SAR Section 3.9.3, "Net Positive Suction Head"

For the above Sections, implementation of MELLLA+ does not change normal reactor operating pressure, decay heat loads, SRV setpoints, or RCIC system hardware, as the system initiation and NPSH requirements are unchanged for operation in the MELLLA+ domain. The RCIC system is further discussed in Section 3.5.3 of this SE.

SAR Section 3.10, "Residual Heat Removal System"

The RHR system is further discussed in Section 3.5.4 of this SE.

SAR Section 3.10.1, "LPCI Mode"

The LPCI mode, as it supports the LOCA response, is discussed in the GGNS SAR, Section 4.2.4, "Low Pressure Coolant Injection."

SAR Section 3.10.2, "Suppression Pool and Containment Spray Cooling Modes"

[[

]].

SAR Section 3.10.3, "Shutdown Cooling Mode"

[[

]], and this system is unaffected by operation in the MELLLA+ domain.

SAR Section 3.10.4, "Steam Condensing Mode"

This topic is not applicable to GGNS.

SAR Section 3.10.5, "Fuel Pool Cooling Assist Mode"

The fuel pool cooling assist mode uses existing RHR heat removal capacity and provides supplemental fuel pool cooling in the event that the fuel pool heat load exceeds the capability of the fuel pool cooling and cleanup system (FPCCS). [[

]]; therefore, operating in the MELLLA+ domain has no effect on the fuel pool cooling assist mode.

SAR Section 3.11, "Reactor Water Cleanup System [RWCU]"; SAR Section 3.11.1, "System Performance"

The licensee concluded that operation in the MELLLA+ domain will not increase the quantity of fission products, corrosion products, and other soluble and insoluble impurities in the reactor water. In addition, there is no significant change in the FW line temperature, pressure, or flow rate. The FW flow rate in the MELLLA+ operating domain decreases slightly from the flow rate in the CLOD. The reactor pressure for the CLOD and in the MELLLA+ operating domain does not change. Therefore, the FW system resistance and operating conditions do not change, and the pressure at the RWCU/FW system interface does not change. The reactor and recirculation system parameters are bounded by, or unchanged, from CLTP conditions. Therefore, there is no adverse effect on the RWCU inlet conditions due to MELLLA+. Because there is no change to the pressure or fluid thermal conditions experienced by the RWCU system, and because there is no increase in the quantity of fission products, corrosion products, and other soluble and insoluble impurities in the reactor water, the implementation of MELLLA+ has no effect on the RWCU system.

SAR Section 3.11.2, "Containment Isolation"

For GGNS, there is no significant change in the FW line temperature, pressure, or flow rate. FW line temperature for the CLOD and in the MELLLA+ operating domain is 420.0 °F (upstream of the RWCU return). The FW flow rate in the MELLLA+ operating domain decreases slightly from the maximum flow rate in the CLOD. As such, the FW flow rates in the MELLLA+ operating domain remain within the FW flow rates in the CLOD. The reactor pressure for the CLOD and in the MELLLA+ operating domain does not change. As such, the FW system resistance and operating conditions do not change, and the pressure at the RWCU/FW system interface does not change for the RWCU return lines. **[[**

]].

3.2.3. SAR Section 4.0, "Engineered Safety Features"

SAR Section 4.1, "Containment System Performance"; SAR Section 4.1.1.1, "Long-Term Suppression Pool Cooling Temperature Response"

Implementation of MELLLA+ does not impact or have an effect on the sensible and decay heat. Thus, **[[** **]]** as a result of operating in the MELLLA+ domain.

SAR Section 4.1.2.3, "SRV Piping - Containment Dynamic Loads" and SAR Section 4.1.2.4, "SRV Containment Dynamic Loads"

The SRV discharge containment loads depend on the SRV setpoints, reactor sensible and decay heat, and long-term SP temperature response. In the GGNS SAR, Sections 4.1.2.3 and

4.1.2.4, the licensee provided an evaluation of the GGNS-specific containment dynamic loads due to SRV discharge. The licensee stated that because the SRV setpoints, **[[ ]]**, and long-term SP temperature response do not change from the current operating domain to the MELLLA+ operating domain, the containment loads due to SRV discharge are unaffected.

<u>SAR Section 4.1.3, "Containment Isolation"; SAR Section 4.1.4, "Generic Letter 89-10", (Reference 47); SAR Section 4.1.6, "Generic Letter 95-07" (Reference 48); and SAR Section 4.1.7, "Generic Letter 96-06" (Reference 49)</u>

The containment pressure and temperature response, which was performed by the licensee, as addressed in SAR Section 4.1.1, "Short-Term Pressure and Temperature Response," for the MELLLA+ domain, confirmed that the MELLLA+ temperature and pressure are bounded by the current licensing basis. As such, operating in the MELLLA+ domain has no effect on any of the above topics.

<u>SAR Section 4.1.5, "Generic Letter 89-16" (Reference 50)</u>

GL 89-16 was issued to address potential severe accident vulnerabilities regarding BWR Mark I containments. GGNS is a Mark III design and therefore, the GL is not applicable to GGNS.

<u>SAR Section 4.2, "Emergency Core Cooling Systems"</u>

<u>SAR Section 4.2.1, "High Pressure Coolant Injection"</u>

This system does not exist at GGNS.

<u>SAR Section 4.2.2, "High Pressure Core Spray"</u>

There is no change in the reactor operating pressure, decay heat, and the SRV setpoints as a result of operating in the MELLLA+ domain. **[[ ]]**. Because HPCS is a part of the LOCA analysis models, the adequacy of the HPCS system performance is confirmed by the analysis.

<u>SAR Section 4.2.3, "Low Pressure Core Spray"</u>

There is no change to the reactor pressure as a result of operating in the MELLLA+ domain. In addition, **[[ ]]**. The adequacy of the LPCS system performance is confirmed by the LOCA analysis.

SAR Section 4.2.4, "Low Pressure Coolant Injection"

There is no change to the reactor pressure as a result of the MELLLA+ operating domain expansion. In addition, [[

]]. The adequacy of the LPCI system performance is confirmed by the LOCA analysis.

SAR Section 4.2.5, "Automatic Depressurization System"

[[

]].

SAR Section 4.2.6, "ECCS Net Positive Suction Head"

The ECCS and containment heat removal pumps are the RHR and core spray (CS) pumps. MELLLA+ does not result in an increase core power and decay heat or the heat addition to the SP during a LOCA, ATWS, SBO, or Appendix R to 10 CFR Part 50 fire event. The licensee stated that there are no physical changes in the piping arrangement that could impact the NPSH analysis. The GGNS does not credit containment accident pressure for calculating the available NPSH for the ECCS pumps, and the SP temperature does not exceed the saturation temperature of 212 °F at atmospheric pressure. Therefore, operation in the MELLLA+ domain does not affect the available NPSH for the ECCS pumps.

SAR Section 4.3, "Emergency Core Cooling Performance"

SAR Section 4.3.1, "Break Spectrum Response and Limiting Single Failure"

The MELLLA+ operating domain expansion will not affect the break spectrum and identification of the limiting break. [[

]].

SAR Section 4.3.4, "Local Cladding Oxidation"

For the GGNS SAR, Sections 4.3.2 and 4.3.3, these show acceptable peak clad temperature (PCT) results that meet the 2,200 °F limit. The margin to the 2,200 °F PCT limit is 470 °F. Note that the local cladding oxidation has been elevated for the limiting large break, as discussed in the SAR, Section 4.3.2, and the resulting value was found to be acceptable. This ensures that the 10 CFR 50.46 limit of 17 percent regarding local cladding oxidation is met for operation in the MELLLA+ domain. PCT is discussed in Sections 3.4.1 and 3.7 of this SE.

SAR Section 4.3.5, "Core-Wide Metal Water Reaction"

The margin to the 2,200 °F PCT limit is 470 °F. The core-wide metal water reaction has been calculated for the limiting large break, discussed in the GGNS SAR, Section 4.3.2, and the resulting value is < 0.1 percent. This ensures that the 10 CFR 50.46 limits regarding core-wide metal water reaction are met for operation in the MELLLA+ domain. As discussed above, PCT is also discussed in Section 3.4.1 of this SE.

SAR Section 4.3.6, "Coolable Geometry" and SAR Section 4.3.7, "Long-Term Cooling"

Compliance with the coolable geometry and long-term cooling acceptance criteria were demonstrated generically for GE BWRs. The MELLLA+ operating domain expansion does not affect the basis for these generic dispositions. These topics are also discussed in Section 3.1.1 of this SE.

SAR Section 4.3.8, "Flow Mismatch Limits"

For most plants, the limits on flow mismatch are more relaxed at lower CF rates. The drive flow mismatch affects the CF coastdown following the break. The effect of the drive flow mismatch on the LOCA evaluation is similar to a small change in the initial CF. The generic MELLLA+ sensitivity studies show that varying the initial CF over a wide range results in a small change in PCT. The effect of the drive flow mismatch is small in comparison to the CF range evaluated for MELLLA+. Because the drive flow mismatch is small, as compared to MELLLA+, the PCT change due to the drive flow mismatch is expected to be smaller than the MELLLA+ sensitivity. Therefore, the current recirculation drive flow mismatch limits remain acceptable in the MELLLA+ domain.

SAR Section 4.4, "Main Control Room Atmosphere Control System"

The main control room atmosphere control system under the MELLLA+ operating domain is affected by the increase in the radiation source term. The licensee stated that the MELLLA+ operating domain expansion does not result in a change in the source term or the release rate, except for a slight increase from the liquid radwaste tank failure. Releases from the liquid radwaste tank failure increase slightly. The GGNS SAR states that the operator exposure from this accident was evaluated and found to be within regulatory limits, with no change to the main control room atmosphere control system. As such, the licensee concluded that operating in the MELLLA+ domain has no effect on this system.

SAR Section 4.5, "Standby Gas Treatment System"; SAR Section 4.5.1, "Flow Capacity"; and SAR Section 4.5.2, "Iodine Removal Capability"

The standby gas treatment system (SBGTS) maintains the secondary containment at a negative pressure and filters the exhaust air by removing fission products present during abnormal conditions. In the SAR, Section 4.5, the licensee provided its evaluation of the SBGTS in the MELLLA+ operating domain. The areas of the SBGTS review include its flow capacity and iodine removal capability. The licensee stated that the SBGTS flow is not affected in the CLTP MELLLA+ operating domain, because the primary and secondary containment leak rates, which depend on the long-term containment pressure response, are unaffected. Therefore, the SBGTS required flow to maintain the required negative pressure in the secondary containment during accident conditions remains unchanged. The licensee stated that the iodine removal capacity of the SBGTS is unaffected, because the core fission product inventory is not changed. As such, there is no change in the absorber iodine loading, decay heat, or iodine removal efficiency while operating in the CLTP MELLLA+ domain.

SAR Section 4.6, "Main Steam Isolation Valve Leakage Control System"

During an accident, the MSIV leakage control system (LCS) controls the release of fission products that would leak through the MSIVs and directs it to the SBGTS. The licensee stated, that the conditions in the steam lines and in the containment following a LOCA are not changed significantly" during MELLLA+ operation. Therefore, the licensee concluded that the MSL collection system capability is unaffected by operating in the MELLLA+ domain.

SAR Section 4.7, "Post-LOCA Combustible Gas Control System"

In the GGNS SAR, Section 4.7, the licensee provided an evaluation of the post-LOCA combustible gas control system. The NRC revised 10 CFR 50.44, "Combustible gas control for nuclear power reactors," in September 2003. The revised rule eliminated the requirements for hydrogen recombiners and relaxed the requirements for hydrogen and oxygen monitoring in containment. The revised 10 CFR 50.44 no longer defines a design-basis LOCA hydrogen release and eliminates requirements for hydrogen recombiners to mitigate such a release. The licensee has adopted the revised rule per License Amendment No. 166, issued in June 2004 (Reference 51), but it has chosen to maintain and leave the recombiners functional. The licensee stated that the MELLLA+ operating domain does not affect the current combustible gas control system. **[[**
]]. As such, the GGNS combustible gas control system will maintain the post-LOCA concentration of oxygen or hydrogen in the containment atmosphere below the lower flammability limit.

3.2.4. SAR Section 5.0, "Instrumentation and Control"

SAR Section 5.1.1, "Average Power Range, Intermediate Range, and Source Range Monitors"

The average power range monitor (APRM) output signals are calibrated to read 100 percent at the CLTP. There is no change in GGNS core power as a result of the MELLLA+ operating domain expansion. [[

]]. The APRMs, IRMs, and SRMs are installed at GGNS in accordance with the requirements established by the GEH design specifications. The GGNS uses normal plant procedures to adjust the IRMs to ensure adequate overlap with the SRMs and APRMs." [[
]]

SAR Section 5.1.2, "Local Power Range Monitors"

There is no change in the neutron flux experienced by the GGNS local power range monitors (LPRMs) resulting from operating in the MELLLA+ domain. As such, the operability, neutronic life, and accuracy of the LPRM detectors are unaffected by operating in the MELLLA+ domain.

SAR Section 5.1.3, "Rod Block Monitors"

This function is not applicable to GGNS.

SAR Section 5.1.4, "Rod Control and Information System"

The rod control and information system (RCIS) supports the operator in making control rod movements. The RCIS provides rod position information to the operator and limits rod movements to ensure that fuel design limits are not exceeded. The rod pattern controller (RPC) and rod withdrawal limiter (RWL) are functions of the RCIS. The low power setpoint (LPSP) is the point at which rod control makes the transition between RPC and RWL control. The LPSP has upper and lower bounding analytical limits. The high power setpoint (HPSP) is the point where the RWL changes allowable control rod withdrawal distances. As such, the implementation of the MELLLA+ domain does not affect the LPSP upper and lower analytical limits or the high power setpoint.

SAR Section 5.2, "BOP Monitoring and Control"

For SAR Sections 5.2.1 through 5.2.6, [[
]], there are no safety or nonsafety-related changes to these systems resulting from the implementation of MELLLA+.

SAR Section 5.3, "Technical Specification Instrument Setpoints"

SAR Section 5.3.2, "Rod Block Monitor"

This is not applicable to GGNS.

3.2.5. SAR Section 6.0, "Electrical Power and Auxiliary Systems"

SAR Section 6.1, "AC [Alternating Current] Power"

Implementing MELLLA+ does not change the GGNS reactor thermal power or the electrical output from the station. As such, **[[**

]]. Therefore, the AC power system is unaffected by operating in the MELLLA+ domain, **[[**

]], and the current emergency power system remains adequate.

SAR Section 6.2, "DC [Direct Current] Power"

Operating in the MELLLA+ domain expansion does not change system requirements for control or motive power loads. As such, there are no changes in the DC power requirements as a result of operating in the MELLLA+ domain.

SAR Section 6.3, "Fuel Pool"

SAR Section 6.3.2, "Crud Activity and Corrosion Products"

Crud activity and corrosion products associated with the GGNS spent fuel do not change due to operating in the MELLLA+ domain.

SAR Section 6.3.3, "Radiation Levels"

The normal radiation levels around the GGNS spent fuel pool (SFP) do not change due to operating in the MELLLA+ domain.

SAR Section 6.4, "Water Systems"

Operating in the MELLLA+ **[[**

]] domain. Therefore, implementation of MELLLA+ does not affect the performance of the safety-related service water systems during and following the most limiting design-basis event, the LOCA. **[[**

]], and implementation of MELLLA+ does not affect the operation of the GGNS service water systems.

SAR Section 6.5, "Standby Liquid Control System"

SAR Section 6.5.1, "Shutdown Margin"

The SLCS shutdown margin for GGNS is calculated as a part of the standard reload process. Because no new fuel product line designs are introduced for the MELLLA+ operating domain expansion, the UFSAR, Section 9.3.5, limit for the minimum SLCS natural boron equivalent concentration of 780 parts per million (ppm) does not change as a result of operating in the MELLLA+ domain. The SLCS is also discussed in Section 3.5 of this SE.

SAR Section 6.6, "Heating, Ventilation, and Air Conditioning"

For GGNS heating, ventilating, and air conditioning (HVAC) systems, the process temperatures and heat load from motors and cables are bounded by the CLTP process temperatures and heat loads; therefore, they are within the design of the HVAC equipment chosen for the worst case conditions. Thus, **[[**

]], and implementation of MELLLA+ has no effect on these systems.

SAR Section 6.7, "Fire Protection"

Implementation of the MELLLA+ does not change the decay heat, reactor vessel water level response, or operator response times because:

- **[[** **]]**;
- **[[** **]]**; and
- **[[** **]]**.

The effect of the MELLLA+ operating domain expansion on PCTs was evaluated by the licensee and found to be acceptable. The effect of the MELLLA+ operating domain expansion on peak SP temperatures and containment pressure response were evaluated in the SAR, Section 4.1, and found to be acceptable. Therefore, the MELLLA+ operating domain expansion does not affect any features of the fire protection design.

SAR Section 6.8, "Other Systems Affected"

The licensee performed a review to assure that the SAR included all systems that may be affected by the implementation on MELLLA+. The licensee has confirmed that those systems that are significantly affected by the operating in the MELLLA+ domain are addressed in the SAR.

3.2.6. SAR Section 7.0, "Power Conversion Systems"

SAR Section 7.1, "Turbine-Generator"; SAR Section 7.2, "Condenser and Steam Jet Air Ejectors"; SAR Section 7.3, "Turbine Steam Bypass"; SAR Section 7.4, "Feedwater and Condensate Systems"

Implementation of MELLLA+ does not change the pressure, steam, FW flowrates, or FW fluid temperature ranges; therefore, the power conversion systems are unaffected by operation in the MELLLA+ domain.

3.2.7. SAR Section 8.0, "Radwaste Systems and Radiation Sources"

SAR Section 8.2, "Gaseous Waste Management"

SAR Section 8.2.1, "Off-Site Release Rate"

The GGNS radiological release rate is administratively controlled to remain within existing release rate limits. In addition, none of the applicable identified parameters are affected by the MELLLA+ operating domain expansion. Therefore, the GGNS offsite release rate relative to the GGNS offgas system is unaffected by the MELLLA+ operating domain expansion.

SAR Section 8.2.2, "Recombiner Performance"

[[

]]. The GGNS-specific value for radiolytic gas flow rate is 0.044 cubic feet per minute/MWt, which does not change as a result of operating in the MELLLA+ domain. Therefore, recombiner performance is unaffected by the MELLLA+ operating domain expansion, and the GGNS recombiner performance is unaffected by the MELLLA+ operating domain.

SAR Section 8.3, "Radiation Sources in the Reactor Core"

The reactor power does not increase as a result of the MELLLA+ operating domain expansion. The GGNS core average exposure for MELLLA+ is **[[**

]], the source terms are unaffected.

SAR Section 8.6, "Normal Operation Off-Site Doses"

SAR Section 8.6.2, "Gamma Shine from the Turbine"

[[

]]. In addition, the increased moisture content in the reactor steam for MELLLA+ operation will not significantly affect the Nitrogen-16 (N-16) activity concentration (in units of microcuries per gram (μCi/g)), because the total N-16 amount contained in the moisture is small compared to that contained in the dry steam. As such, operating in the MELLLA+ domain does not affect the gamma shine from the turbine.

3.2.8. SAR Section 9.0, "Reactor Safety Performance Evaluations"

SAR Section 9.1, "Anticipated Operational Occurrences"

AOOs are discussed in Section 3.6 of this SE.

SAR Section 9.1.3, "Non-Limiting Events"

The licensee reviewed each of the GESTAR (see SRP Reference 35(q)) limiting AOO events and key non-limiting events and determined that there is no effect on these events by operating in the MELLLA+ domain.

SAR Section 9.2, "Design Basis Accidents and Events of Radiological Consequence"

SAR Section 9.2.1.1, "Control Rod Drop Accident"

For GGNS, the postulated control rod drop accident (CRDA) event involves a sudden control rod drop from the core, resulting in the failure of 16 fuel bundles and the release of noble gases, halogens, and alkali metals in the melted/failed fuel into the reactor coolant system (RCS). The release path is via the condenser. The CRDA release is dependent on the source terms and maximum peaking factor. Operation in the MELLLA+ operating domain does not affect the CRDA source terms, and the peaking factor remains bounding. There are no changes to the removal, transport, or dose conversion assumptions for this event. Therefore, the GGNS CRDA evaluation for the MELLLA+ operating domain is bounded by the analysis for the current licensing basis.

SAR Section 9.2.1.2, "Main Steam Line Break Accident (Outside Containment)"

For GGNS, the source terms for the MSL break accident are dependent on the relative amount of water and steam released. There are no changes to the removal, transport, or dose conversion assumptions for this event. Radionuclide concentrations are set at conservative values for the coolant source terms and at TS limits, which remain bounding and unchanged.

The MELLLA+ operating domain expansion results in more steam voids in the reactor vessel, resulting in a larger fraction of steam release than in the CLOD. The fission product release is weighted by the water because the concentration of iodine in water is approximately 45 times that of steam. The increase in steam and decrease in water results in lower releases such that the current analysis is bounding. Therefore, the GGNS MSL break accident evaluation is not affected by the MELLLA+ operating domain expansion.

SAR Section 9.2.1.3, "Loss-of-Coolant Accident (Inside Containment)"

The design input and assumptions for SP pH were previously evaluated. The source term assumptions and the acid production terms do not change as result of operating in the MELLLA+ domain. The use of sodium pentaborate as a buffer per GGNS UFSAR, Section 15.6.5.5.2, continues to be appropriate. **[[**

]]; therefore, the GGNS LOCA evaluation is not affected by operating in the MELLLA+ domain.

SAR Section 9.2.1.5, "Fuel Handling Accident"

[[

]]. Therefore, the GGNS FHA evaluation for the MELLLA+ operating domain is bounded by the analysis for the CLOD.

SAR Section 9.2.1.6, "Offgas System Failure"

[[

]]. Therefore, the GGNS offgas system failure evaluation is not affected by operating in the MELLLA+ domain.

SAR Section 9.2.1.7, "Additional Review Areas"

A temporary departure from nucleate boiling and the subsequent assumption of fuel failure may occur for infrequent events. This fuel failure is currently assumed for the pressure controller failure (conservatively estimated to be core-wide), misplaced fuel bundle (five bundles - the misplaced bundle and the four surrounding bundles), and the recirculation pump seizure event (conservatively estimated to be core-wide). The source terms for the above accidents are based on the gap inventory and peaking factors as applicable. There are no changes to the removal, transport, or dose conversion assumptions for these events, and the rod inventories are not changed and the peaking factor remains bounding. Therefore, the GGNS evaluation for these events at the MELLLA+ operating domain is bounded by the analyses for the CLOD.

The MSIV closure event is analyzed based on reactor coolant and steam releases, assuming maximum iodine spiking permitted by the plant's TSs, and is unaffected by operating in the MELLLA+ domain. There are no changes to the removal, transport, or dose conversion assumptions for this event. Therefore, the GGNS MSIV closure evaluation is not affected by operating in the MELLLA+ domain.

SAR Section 9.2.2, "Other Events with Radiological Consequences"

SAR Section 9.2.2.1, "Instrument Line Break Accident"

This line break is not applicable to GGNS.

SAR Section 9.2.2.2, "Large Line Break (Feedwater or Reactor Water Cleanup)"

In accordance with the GGNS licensing basis, the dose consequences of a Large Line Break are enveloped by the MSL break. In addition, coolant concentrations are based on TS levels. Mass flows are based upon TS valve closure time and flow rates, which are unaffected by operating in the MELLLA+ domain. Therefore, the dose consequences of a large line break are not changed under MELLLA+ and bounded by the MSL break.

SAR Section 9.2.2.3, "Cask Drop"

This event is not applicable to GGNS, as the spent fuel storage cask crane is prohibited from travelling over the SFP.

SAR Section 9.3, "Special Events"

SAR Section 9.3.2, "Station Blackout"

Implementing MELLLA+ does not change in the reactor power level or the decay heat or reactor operating pressure. For GGNS, there are no significant changes in the MS flow rate. Therefore, the GGNS response to and coping capabilities for the SBO event are not affected by operation in the MELLLA+ domain.

3.2.9. SAR Section 10.0, "Other Evaluations"

SAR Section 10.1, "High Energy Line Break"

SAR Section 10.1.1, "Steam Lines"

The licensee stated that a review of the heat balances produced for operating in the MELLLA+ domain confirmed there is no effect on the steam pressure or enthalpy at the postulated break locations (e.g., MS and RCIC). Therefore, operation in the MELLLA+ domain has no effect on the mass and energy releases from a high energy line break in a steam line.

SAR Section 10.1.2, "Balance-of-Plant Liquid Lines"

The licensee stated that a review of the heat balances produced for operating in the MELLLA+ domain confirmed there is no effect on the liquid line conditions at the postulated FW, RWCU, and RHR break locations. In addition, the mass and energy release for operation in the MELLLA+ domain is bounded by the MELLLA domain analyzed for EPU, including FWHOOS. Therefore, MELLLA+ has no adverse effect on the existing mass and energy releases from a high energy line break in a FW, RWCU, or RHR line.

SAR Section 10.1.3, "Other Liquid Lines"

The licensee stated that a review of the heat balances produced for operating in the MELLLA+ domain confirmed there is no effect on the liquid line conditions (excluding FW, which is addressed in the GGNS SAR, Section 10.1.2) at the postulated break locations. **[[**

]]. The scope of these evaluations include the MELLLA+ operating domain expansion effects on subcompartment pressures and temperatures, pipe whip, jet impingement, and flooding, consistent with the plant licensing basis.

SAR Section 10.2, "Moderate Energy Line Break"

SAR Section 10.2.1, "Flooding"

The licensee stated that a review of the GGNS auxiliary flow rates and system inventories shows the MELLLA+ operating domain expansion does not affect the flow rates of moderate energy piping systems. In addition, no operational modes evaluated for moderate energy line break (MELB) are affected by the MELLLA+ operating domain expansion. Therefore, the GGNS internal flooding analysis and safe shutdown analysis are not affected by operating in the MELLLA+ domain.

SAR Section 10.2.2, "Environmental Qualification"

The licensee stated that a review of the GGNS auxiliary flow rates and system inventories shows that operating in the MELLLA+ domain does not affect the flow rates of moderate energy piping systems. Also, for GGNS, no operational modes evaluated for MELB are affected by operating in the MELLLA+ domain. Therefore, the licensee concludes that the GGNS internal flooding analysis and safe shutdown analysis and MELB analysis are not affected by operating in the MELLLA+ domain.

SAR Section 10.3, "Environmental Qualification"; SAR Section 10.3.1, "Electrical Equipment"

Implementation of MELLLA+ does not change the reactor power radiation levels in any of the plant areas where safety-related equipment is located. The licensee has **[[**

]]. The licensee also concludes [[ ]]. Therefore, the licensee concluded that there is no change to the environmental qualification (EQ) for safety-related electrical equipment located inside or outside of containment as result of operating in the MELLLA+ domain.

SAR Section 10.3.2, "Mechanical Equipment With Non-Metallic Components"

Implementing MELLLA+ does not change the normal process temperatures or [[ ]]. Therefore, there is no change to the EQ for safety-related mechanical equipment with non-metallic components located inside or outside of containment as result of operating in the MELLLA+ domain.

SAR Section 10.3.3, "Mechanical Component Design Qualification"

Implementation of MELLLA+ does not change normal process temperatures, pressures, and flow rates. In addition, there is no change in radiation levels in any of the plant areas where safety-related equipment is located. [[ ]]. Therefore, the mechanical components and component supports are adequately designed for the MELLLA+ operating domain.

SAR Section 10.7, "Plant Life"; SAR Section 10.7.2, "Flow-Accelerated Corrosion"

For GGNS, the evaluation of and inspection for flow-induced erosion/corrosion in piping systems affected by flow-accelerated corrosion (FAC) is addressed by compliance with NRC GL 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning" (Reference 52). The requirements of GL 89-08 are implemented at GGNS by utilization of the Electric Power Research Institute (EPRI) generic program, CHECWORKSTM. GGNS-specific parameters are entered into this program to develop requirements for monitoring and maintenance of specific system components. The FAC monitoring programs are adequate to manage potential effects of the MELLLA+ operating domain expansion.

For GGNS, there are no significant changes in MS or FW temperatures or MS or FW flow rates in the MELLLA+ operating domain compared to the CLOD. As previously discussed, there is a small increase in average moisture content of the MS leaving the reactor during short periods of the cycle. In addition to the MELLLA+ startup testing described in the GGNS SAR, Section 10.4.1, "Steam Separator-Dryer Performance," GGNS routinely monitors the moisture content of reactor steam. Any increases in moisture carry-over (MCO) above the current design limit of 0.1 weight (wt) percent will be evaluated for the effect on the FAC monitoring program. Therefore, there is no change in the potential for unrecognized erosion/corrosion due to FAC for operating in the MELLLA+ domain. Entergy has made two regulatory commitments to test and determine the MCO value at CLTP and MELLLA+ conditions. This data will be used to determine the effect of the MCO on the FAC monitoring program and for defining the MCO magnitude and trend.

SAR Section 10.8, "NRC and Industry Communications"

The evaluations and calculations included in this MELLLA+ SAR, along with any supplements, demonstrate that operating in the MELLLA+ domain can be accomplished within the applicable design criteria. Because these evaluations of plant design and safety analyses inherently include any effect as a result of NRC and industry communications, the NRC concludes it is not necessary to review prior communications.

3.2.10 Conclusion

The NRC staff concludes that for the generic dispositions discussed in this SE, Sections 3.2.1 through 3.2.9, operation in the MELLLA+ domain does not change the operation of the GGNS from its current licensing basis. In general, this conclusion was based on several limitations or conditions of operating in the MELLLA+ domain, including the following:

- Reactor power level unchanged.
- Fuel design unchanged.
- Reactor temperature and pressure unchanged.
- Design and accident containment pressure and temperature and SP temperature unchanged.
- Reduced core flow in the MELLLA+ domain.
- No change to safety and BOP system hardware or capability.
- Operating temperatures, pressures, and flows for safety and BOP systems unchanged.
- Sensitivity to MELLLA+ is small enough that the required plant cycle-specific reload analysis process is sufficient and appropriate for establishing the MELLLA+ licensing basis.
- Approximately the same decay head loads for the core and SFP (increase by 1 percent for the SFP).

Based on the above, the NRC staff concludes that the above topics have been dispositioned by one of the four criteria proposed by the licensee, as noted at the beginning of Section 3.2 of this SE.

3.3. Plant-Specific Dispositions

For the following topics, a plant-specific review was performed to disposition the issue.

SAR Section 3.1.2, "Overpressure Relief Capacity"

The plant-specific evaluation concludes that for GGNS, the limiting overpressure event is the MSIV closure with scram on high-flux. The peak RPV bottom head pressure is unchanged and remains less than the ASME limit of 1375 pounds per square inch gauge (psig) for AOOs. The

SRV tolerance assumed in the GGNS ASME overpressure event analysis is 3 percent. For non-AOOs, the ATWS analysis concludes that overpressure relief capacity is adequate. The analysis of the limiting overpressure event for GGNS demonstrates that no change in overpressure relief capacity is required. The ATWS analysis concludes that no increase in the number of SRVs credited in the analysis is required to demonstrate acceptable results. As such, no other changes in the pressure relief system or SRV setpoints are required for MELLLA+. The plant-specific analysis of the overpressure relief capacity topic finds that operation in the MELLLA+ domain has no effect on the GGNS overpressure relief capacity or SRV setpoint tolerances. The ASME overpressure event continues to be analyzed as a part of each reload analysis and is reported in the SRLR. Therefore, the NRC staff concludes that as the peak vessel pressure remains unchanged and is below the ASME, Section III limit of 1375 psig limit, operation in the MELLLA+ domain does not change the needed overpressure relief capacity. Systems providing overpressure protection for the RCPB are also discussed in Section 3.5.2 of the SE.

SAR Section 3.2.1, "Fracture Toughness"

MELLLA+ operation results in slightly higher fluxes at the vessel because of the reduced moderation. By letter dated August 18, 2015 (ADAMS Accession No. ML15229A218), Amendment No. 204, the NRC staff approved a revised fluence methodology for GGNS. This methodology will used in the Core Operating Limits Report to maintain the pressure-temperature curves to address fracture toughness.

SAR Section 3.3.1.3, "Reactor Internal Pressure Differences (Acoustic and Flow-Induced Loads) for Faulted Conditions"

The plant-specific LOCA analysis concludes that the loads in the RPV annulus on the jet pumps, core shroud, and core shroud support are not increased as a result of the MELLLA+ operating domain expansion. As such, the NRC staff concludes that operating in the MELLLA+ domain does not affect these loads.

SAR Section 3.3.2, "Reactor Internals Structural Evaluation"

[[

]]. As such, the NRC staff concludes that operating in the MELLLA+ domain has no effect on the reactor structural evaluations.

SAR Section 3.3.3, "Steam Separator and Dryer Performance" and SAR Section 3.3.4, "Steam Line Moisture Performance Specification"

The analysis indicates that there will be a small (~0.1 wt percent) increase in moisture in the separators when operating in the MELLLA+ domain, but it is within acceptable steam separator-dryer performance. The MCO for GGNS may increase to a maximum of 0.33 wt percent during the cycle when GGNS is operating at or near the MELLLA+ minimum CF rate.

The separator moisture content is monitored during operation to ensure it remains within the acceptable limit of 0.33 wt percent. The design MELLLA+ evaluations assumed a conservative value of 0.35 percent for MCO.

Regulatory Evaluation

Plant operation at EPU conditions or at combined EPU and MELLLA+ conditions can result in adverse flow effects on the MS, FW, and condensate systems and their components (including the steam dryer in BWR plants) from increased system flow and FIV. Some plant components, such as the steam dryer, do not perform a safety function, but must retain their structural integrity to avoid the generation of loose parts that might adversely impact the capability of other plant equipment to perform their safety functions. As part of the EPU review, the NRC staff reviewed an evaluation by Entergy, of potential adverse flow effects for the proposed EPU operation at GGNS. This review included consideration of the design input parameters, design-basis loads, and load combinations for the GGNS steam dryer for normal operation, upset, emergency, and faulted conditions. The NRC staff's review covered the analytical methodologies, assumptions, and computer modeling used in the evaluation of the GGNS replacement steam dryer. The NRC staff's review also included a comparison of the resulting stresses against applicable limits.

The NRC staff reviewed the licensee's evaluation of the steam dryer components at GGNS for potential susceptibility to adverse flow effects from EPU and MELLLA+operation. The NRC's acceptance criteria is based on the GDC in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR 50 including (1) GDC 1 insofar as it requires that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC 2 insofar as it requires that structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions, and (3) GDC 4 insofar as it requires that structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. The NRC SRP (Reference 35), Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.5, contain specific review criteria regarding the adverse flow effects. In its review, the staff also utilized RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Startup Testing" (Reference 37).

TECHNICAL EVALUATION

Licensee Input

By letter dated August 26, 2014 (Reference 7), the licensee provided a response to RAI-21 regarding the steam dryer's acceptability for the combined MELLLA+ and EPU conditions. This response included a summary of its evaluations. The evaluations considered the effect of MELLLA+ conditions on MCO, acoustic boundary conditions, steam dryer loads, and steam dryer stresses. The licensee stated that MELLLA+ core flow conditions affect the steam quality at all locations in the steam dryer/steam separator system. The highest MCO predicted under MELLLA+ conditions is less than 0.20 wt percent. The lower core flow conditions at MELLLA+ affect the acoustic boundary conditions for the plant-based load evaluation model and acoustic properties, namely acoustic damping and speed of sound, which determine the acoustic loading on the steam dryer. MELLLA+ operation also increases the moisture entrained in the steam both upstream and downstream of the steam dryer. The licensee also evaluated the steam dryer stresses for the combined EPU and MELLLA+ flow conditions, comparing them with the acceptance criteria in the ASME Boiler and Pressure Vessel Code (ASME Code) (Reference 63).

NRC Staff Evaluation

The NRC staff reviewed the licensee's steam dryer evaluations. The licensee conservatively used a high bounding MCO of 0.35 wt percent for the steam exiting the RPV in the MELLLA+ evaluations. However, in a subsequent evaluation for the steam line components, the licensee determined that the bounding MCO value for these components is 0.33 wt percent MCO. The licensee stated it will be using 0.33 wt percent as the bounding value for any future MCO evaluations. As all other components analyses were performed at 0.35 wt percent no reanalyses were required (i.e., 0.33 wt percent evaluations are bounded by any analysis performed at .0.35 wt percent). As noted above, since the highest MCO predicted under MELLLA+ conditions is less than 0.20 wt percent both these values are conservative.

The effect of acoustic properties corresponding to MELLLA+ conditions is insignificant on the acoustic pressure loads on the steam dryer, except for the inner hood region. Based on sensitivity studies, the licensee demonstrated that with conservative assumptions, there is adequate structural margin to accommodate any increases in acoustic loads in the inner hood region for MELLLA+ operating conditions.

The MELLLA+ CF conditions also affect the steam dryer stresses. The CF effects primarily impact lower dryer components below the support ring. The licensee developed a correlation to characterize the CF effects on steam dryer stresses, and demonstrated that the lower CF at MELLLA+ resulted in a small increase in stresses in dryer components. The licensee showed that the minimum alternating stress ratio at the limiting location, located at the dryer skirt Tee changed from 1.38 at EPU conditions to 1.27 at the combined EPU and MELLLA+, due to a small increase in FIV stress. This is still higher than the lowest acceptable limit of 1.0; therefore, the dryer would maintain its structural integrity while operating in the MELLLA+ domain.

In addition to FIV stresses, the licensee also evaluated the steam dryer stresses for the combined effect of MELLLA+ and EPU conditions for the ASME normal, upset, emergency, and faulted load combinations, and demonstrated that the respective ASME Code based allowable stress limits are met.

Conclusion

The NRC staff reviewed the licensee's steam dryer evaluations for the combined MELLLA+ and EPU conditions and found them acceptable. The stresses in the steam dryer meet the acceptance criteria for the FIV stress limit (13600 psi endurance limit for high cycle fatigue).

The NRC staff concludes that the proposed license amendment to operate GGNS at the proposed EPU conditions, combined with MELLLA+ for the steam dryer, is acceptable with respect to potential adverse flow effects for high cycle fatigue, as well as the ability to withstand the ASME normal, upset, emergency, and faulted load combinations. The NRC staff also concludes that the steam dryer will maintain its structural integrity for the combined EPU and MELLLA+ flow conditions.

As noted above, the moisture content of the steam leaving the RPV increases in the MELLLA+ domain. The effect of increasing steam moisture content has been analyzed in the tasks that use the MCO value from the GGNS SAR, Sections 3.3.3, "Steam Separator and Dryer Performance," and 3.3.4, "Steam Line Moisture Performance Specification." The effects of increased moisture are also discussed in the following:

SAR Section 3.4.1, "FIV Influence on Piping"
SAR Section 3.5.1, "Reactor Coolant Pressure Boundary Piping"
SAR Section 5.2.4, "Feedwater Control System (Normal Operation)"
SAR Section 8.1, "Liquid and Solid Waste Management"
SAR Section 8.4.2, "Fission and Activated Corrosion Products"
SAR Section 8.5, "Radiation Levels"
SAR Section 8.6.1, "Plant Gaseous Emissions"
SAR Section 10.7.2, "Flow Accelerated Corrosion"

The licensee has made two regulatory commitments, as discussed in Section 4.3 of this SE, to test and determine the MCO value at CLTP and MELLLA+ conditions. This data will be used to determine the effect of the MCO on the FAC monitoring program and for defining the MCO magnitude and trend.

SAR Section 4.0, "Engineered Safety Features"

SAR Section 4.1, "Containment System Performance"

For the review of the SAR Section 4.1, the NRC staff acceptance criteria was based on GDC 4, 16, 19, 38, 41, and 50. In addition the following SRP Sections 6.2.1.1.C, 6.2.1.2, 6.2.5, and 6.2.2 (Reference 35) as supplemented by Regulatory Guide (RG) 1.82 Revision 4, "Water

Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (Reference 53), were considered in the review. The staff also considered the requirements of 10 CFR 50.44, "Combustible gas control for nuclear power reactors," insofar as it requires that plants be provided with the capability of controlling combustible gas concentrations in the containment atmosphere,

SAR Section 4.1.1, "Short-Term Pressure and Temperature Response"

The purpose of the short-term analysis is to confirm that containment peak pressure and temperature do not exceed their design limits with the proposed change. The licensee used LAMB computer code (Reference 54) for the short-term mass and energy release analysis and M3CPT computer code (Reference 55) for the short-term containment pressure and temperature response analysis for the proposed EPU MELLLA+ operating domain, which are the same as those used in the current analysis. The EPU thermal power is the CLTP for GGNS.

In the SAR, Section 4.1.1, the licensee stated that the GGNS short-term recirculation suction line break (RSLB) LOCA drywell and containment pressure and temperature responses are affected by the change in the break enthalpy as a result of the MELLLA+ operating domain expansion. The analysis inputs for the short-term RSLB and main steam line break (MSLB) are consistent with the analysis of record (AOR), with the exception of the wetwell (WW) airspace volume values at short-term pressure low-water level and high-water level, which have been updated. The WW low-water level airspace volume is revised from 139,933 ft^3 to 153,125 ft^3, and the WW high-water level airspace volume is revised from 136,786 ft^3 to 149,978 ft^3. The proposed change results in an increase in the WW airspace volume by 13,192 ft^3. In response to RAI-3 by letter dated August 26, 2014 (Reference 7), the licensee stated that the increase in the free WW volume is due to using a more accurate hydraulic control unit (HCU) floor thickness and affects only the air space volume below the HCU floor. There is no change in the SP volume or level. The containment volume did not change in the MELLLA+ analyses; only the air space volume below the HCU floor increased. The licensee included the HCU floor in a conservative manner in the short-term analyses.

The licensee's calculated short-term peak drywell pressure changed from 27.0 psig in the AOR to 25.2 psig in the proposed analysis; both of which are limiting for the MSLB LOCA case. The licensee's calculated short-term peak containment pressure changed from 14.8 psig to 12.1 psig, both of which are limiting for the MSLB LOCA case. Since the peak containment pressure is used for Appendix A leakage testing, the licensee proposes revising the P_a (calculated peak containment accident pressure) to the new calculated limit of 12.1 psig. The licensee's calculated short-term peak drywell (DW)-to-containment differential pressure changed from 24.2 psi, which is limiting for RSLB LOCA case in the AOR to 23.6 psi, which is limiting for the MSLB LOCA case in the proposed analysis. The calculated values of the peak DW pressure, peak containment pressure, and the peak DW-to-containment pressure in the proposed analysis are below their design limits of 30 psig, 15 psig, and 30 psig, respectively.

Based on the above, the NRC staff concludes that analysis for the short-term peak drywell temperature for the RSLB analysis cases at MELLLA+ demonstrates that it is bounded by the CLTP results reported in GGNS License Amendment No. 191 (Reference 24), which remains below the design limit of 330 °F.

<u>SAR Section 4.1.2, "Containment Dynamic Loads"</u>

By letter dated July 18, 2012, the NRC staff issued License Amendment No. 191 which requires that a plant-specific evaluation be performed to determine the effect of the MELLLA+ operating domain expansion on the LOCA containment dynamic loads. These loads include vent clearing jet loads, pool swell, condensation oscillation, and chugging loads. For GGNS, the LOCA hydrodynamic loads are defined in the UFSAR, Appendices 6A and 6D.

<u>SAR Section 4.1.2.1, "LOCA Loads"</u>

A plant-specific evaluation was performed to determine the effect of MELLLA+ on the LOCA containment dynamic loads. Results from **[[** **]]**, are used to evaluate the effect of the MELLLA+ operating domain expansion on LOCA containment dynamic loads. The LOCA containment dynamic loads include vent clearing jet loads, pool swell, condensation oscillation, and chugging. The results of the plant-specific LOCA containment dynamic loads evaluation demonstrate that existing vent clearing jet loads, pool swell, condensation oscillation, and chugging load definitions remain bounding for operation in the MELLLA+ operating domain. Therefore, the LOCA containment dynamic loads are not affected by the MELLLA+ operating domain expansion. The NRC staff's review of the current containment pressure and temperature response confirmed that it remains bounding. Consequently, the staff concludes that the LOCA containment dynamic loads are not affected by the MELLLA+ operating domain.

<u>SAR Section 4.1.2.2, "Subcompartment Pressurization," "Annulus Pressurization Load Evaluation," and "Subcompartment Pressurization Evaluation"</u>

The subcompartments considered in the evaluation are the biological shield wall (BSW) annulus and the region under the DW head. The subcompartment pressurization loads are affected by the short-term containment response.

The licensee stated that the differential pressure loading on the BSW is not significantly affected by MELLLA+. The peak BSW asymmetric pressure load resulting from the limiting recirculation pump discharge line break in the MELLLA+ operating domain expansion remains below the BSW design differential pressure. The original BSW design used conservative asymmetric and uniform pressure loads. Based on the above, the NRC staff concludes that the licensee's evaluation is acceptable.

The critical item to be considered for DW head subcompartment is the loading on the DW head refueling bulkhead plate due to a postulated break in the RCIC system head spray line in the subcompartment. The licensee stated that the pressure loading on the DW head refueling bulkhead plate, due to a postulated break in the RCIC system head spray line in the drywell head subcompartment, is not affected by the MELLLA+ operating domain expansion. This is because the steam dome pressure is not changed from the current operating domain to the MELLLA+ operating domain. Therefore, the drywell head refueling bulkhead plate pressure loads are not affected by operating in the MELLLA+ operating domain expansion. The EPU was approved using the Constant Pressure Power Uprate methodology, which requires that the dome pressure remain unchanged. As such, the dome pressure must be maintained as a result of implementing the MELLLA+ domain and the NRC staff concludes that the licensee's evaluation is acceptable and therefore, the sub-compartment pressurization loads are not affected by the MELLLA+ operating domain expansion. To change the dome pressure would require a license amendment.

Conclusion for SAR Section 4.1 Topics

The NRC staff reviewed the licensee's assessment of the topics in SAR, Section 4.1, addressed in this SE and concludes that it's adequately addressed for GGNS in the CLTP MELLLA+ operating domain. The staff also concludes that GGNS will continue to meet the requirements of GDC 4, 16, 19, 38, 41, and 50, following implementation of the proposed MELLLA+ operating domain under EPU conditions.

SAR Section 4.3, "Emergency Core Cooling System Performance"

SAR Section 4.3.2, "Large Break Peak Clad Temperature"

The licensee's evaluation concludes that MELLLA+ primarily affects the first PCT; therefore, the limiting single failure is not affected by MELLLA+. The evaluation concludes that PCT performance in GGNS is affected by MELLLA+, but it is within acceptable limits. LOCA analyses are presented to demonstrate the PCT performance. PCT is discussed in Section 3.4.1 of this SE.

SAR Section 4.3.3, "Small Break Peak Clad Temperature"

Small break calculations were not performed at MELLLA+ conditions, because the PCT results at full flow are significantly lower than the current 10 CFR Part 50, Appendix K, PCT limits. PCT is discussed in Section 3.4.1 of this SE.

SAR Section 5.1.5, "Traversing Incore Probes"

Because thermal traversing incore probes (TIPs) are affected by bypass voiding above the D-level LPRMs in excess of 5 percent, operator actions and procedures that mitigate the effect of bypass voiding on the thermal TIPs and the core simulator used to monitor the fuel performance are requested in the MELLLA+ LTR SER Limitation and Condition 12.15 for operation. These items are not required for GGNS, because hot channel bypass voiding at the TIP exit elevation is not in excess of 5 percent for the entire MELLLA+ operating domain.

SAR Section 5.3, "Technical Specification Instrument Setpoints"

The MELLLA+ operating domain expansion results in the development of two allowable values (AVs) (APRM Flow-Based Scram and Rod Block Monitor). GEH uses the approved simplified process to determine the instrument nominal trip setpoints (NTSPs) for MELLLA+ applications. The NRC staff has previously reviewed and accepted the simplified approach in the review of NEDC-33004P-A (Reference 56). Consistent with that approval for GGNS, the following criteria are satisfied for using the simplified process:

1. [[ ]].

2. NRC-approved GEH or plant-specific methodologies are used.

3. [[ ]].

4. [[ ]].

SAR Section 5.3.1, "APRM Flow-Biased Scram"

This function is referred to in the GGNS TSs as the APRM flow-biased simulated thermal power (STP) – high function. The APRM flow-biased STP – high function is not associated with a limiting safety system setting and consequently does not have an analytical limit. The MELLLA+ APRM flow-biased AV expressions are:

- AV_M+ROD BLOCK = 0.64W + 58.8 percent, for the Rod Block
- AV_M+SCRAM = 0.64W + 61.8 percent, for the scram

SLO is not applicable to the MELLLA+ operating domain as discussed in Section 3.6.3 of the SAR. Therefore, for SLO, the flow-biased AVs are unchanged.

The licensee stated the evaluation of APRM flow-biased scram setpoints is consistent with the methods described for plant-specific evaluations of this topic in the MELLLA+ LTR. As such, the NRC staff concludes that the APRM flow-biased scram setpoints for the GGNS plant-specific evaluation are acceptable.

SAR Section 6.3.1, "Fuel Pool Cooling"

For GGNS, the [[

]] is well within the capacity of the existing fuel pool cooling system (FPCS). Therefore, the NRC staff concludes that the FPCS remains capable of meeting its design criteria and is acceptable for the operating in the MELLLA+ domain.

SAR Section 6.3.4, "Fuel Racks"

SAR Section 6.3.4.1, "Spent Fuel Storage Criticality Review"

The wet fuel storage facilities, which include the SFP and upper containment pool (UCP) continue to rely on a neutron absorber (poison) to maintain sub-criticality. GGNS has Boraflex fuel storage racks in both the SFP and UCP. These racks are monitored and evaluated to verify their acceptability. The racks are categorized into two regions based on the program results, which consider dose and boron carbide loss. Each assessment includes projections to confirm acceptable performance through the subsequent evaluation period. This program is described in the GGNS UFSAR, Section 9.1.2.3. Fuel assemblies are evaluated every cycle for storage in wet fuel storage racks. The evaluation confirms the reload fuel is less reactive than the bounding fuel assembly assumed in the criticality safety AOR. The evaluation considers a range of void fraction histories and exposures that cover MELLLA+ operations, and any changes in fuel enrichment, gadolinia, or fuel geometry.

Implementation of MELLLA+ allows the plant to operate at lower CFs. This application will not change the rated core power, no change in fuel, etc., so most of the relevant depletion parameters that affect criticality analyses would not change (operating temperature, power density, etc.). The one exception is the void fraction. With the lower CFs, the maximum void fraction at the top of the core will increase, and this will harden the neutron spectrum and lead to more Pu production.

The licensee has stated that the range of void fractions considered in the criticality analyses already cover the maximum void fraction expected from MELLLA+ operations. As such, the conditions analyzed in the criticality analyses continue to bound MELLLA+ operation. Based on the above, the NRC staff concludes that the criticality safety AOR bounds the MELLLA+ fuel that will be stored in the wet fuel storage racks.

SAR Section 6.5, "Standby Liquid Control System"

The SLCS is discussed in Section 3.5.5 of this SE.

SAR Section 6.5.2, "System Hardware"

The GGNS reactor operating pressure is unchanged by the MELLLA+ operating domain expansion. Thus, there are no changes to the GGNS SRV setpoints as a result of operating in the MELLLA+ domain. Because the reactor dome pressure and SRV setpoints are unchanged for MELLLA+, the SLCS process parameters do not change. Therefore, the capability of the SLCS to perform its shutdown function is not affected by operating in the MELLLA+ domain. **[[**

]]. Thus, the operation of the SLCS pump relief valves has been evaluated for GGNS in the GGNS SAR Section 6.5.3, below.

SAR Section 6.5.3, "ATWS Requirements"

The SLCS ATWS performance is evaluated in the SAR Section 9.3, for a representative core design in the MELLLA+ operating domain. The representative MELLLA+ evaluation shows that the SLCS maintains the capability to mitigate an ATWS and that the current boron injection rate is sufficient relative to the peak SP temperature. The ATWS analysis in the GGNS SAR, Section 9.3.1, also demonstrates that there is no increase in the peak vessel dome pressure during the time the SLCS is in operation. Using the GGNS plant-specific ATWS analysis, the maximum expected SLCS pump discharge pressure for the limiting ATWS event is 1369.3 psig, based on a reactor upper plenum pressure of 1222.3 psig, and an SLCS pressure drop of 147 psi. The pressure margin for the pump discharge relief valves remains above the minimum value needed to ensure that the standby liquid control (SLC) relief valves remain closed during system injection. The minimum reactor pressure just prior to the time when SLCS initiates remains low enough to ensure SLC relief valve closure prior to the analyzed SLCS initiation time in the event of an early initiation of the SLCS during the initial ATWS transient pressure response. Consequently, the current GGNS SLCS pump discharge pressure will be increased to ensure ATWS requirements are met. The NRC staff concludes that as a result of the increased discharge pressure, the TS value for the relief valve setpoint needs to be increased (the licensee had proposed a value of 1370 psig). This is discussed in Section 3.5.5 of this SE.

SAR Section 8.0, "Radwaste Systems and Radiation Sources"; SAR Section 8.1, "Liquid and Solid Waste Management"; SAR Section 8.1.1, "Coolant Fission and Corrosion Product Levels"

A discussion of the coolant activation products, as well as fission and activated corrosion product levels in the coolant, is discussed in the SAR Section 8.4, "Radiation Sources in Reactor Coolant."

SAR Section 8.1.2, "Waste Volumes"

There is no change in the reactor power level or MS or FW flow rates as a result of operating in the MELLLA+ domain. For GGNS, an evaluation was performed using a conservatively high MCO of 0.35 wt percent. This value bounds the expected MCO as a result of operating in the MELLLA+ operating domain. The GGNS evaluation indicated that most of the fission and corrosion products carried over are removed in the moisture separator reheater and returned to the reactor vessel via the FW system. The very small amounts of MCO and fission and corrosion products that pass through the low pressure turbine to the condenser result in a negligible increase in the loading on the condensate full-flow filtration (CFFF) filters and the condensate demineralizers. Due to the very small increase in reactor MCO reaching the condenser, the CFFF filter backwash frequency and volume are not changed, and the disposal frequency of the condensate demineralizer resins is not changed. The NRC staff concludes that since the RWCU filter demineralizer backwash frequency is not changed, the RWCU system is not affected by operation in the MELLLA+ operating domain.

SAR Section 8.4, "Radiation Sources in the Reactor Coolant"

SAR Section 8.4.1, "Coolant Activation Products"

The reactor power and steam flow rate do not change as a result of operating in the MELLLA+ domain. [[ ]], there is no change in the GGNS coolant activation products as a result of operating in the MELLLA+ domain.

SAR Section 8.4.2, "Fission and Activated Corrosion Products"

For GGNS, reactor power, MS and FW flow rates do not change as a result of operating in the MELLLA+ domain. Therefore, the MELLLA+ operating domain expansion does not affect the total activity concentration in the reactor coolant. The moisture content of the MS leaving the vessel has been conservatively assumed to increase up to 0.35 wt percent at times while operating near the minimum CF in the MELLLA+ operating domain. The distribution of the fission and activated corrosion product activity between the reactor water and steam is affected by the increased moisture content. With increased MCO, additional activity is carried over from the reactor water with the steam as discussed in the SAR, Sections 3.3.3 and 3.3.4.

SAR Section 8.5.1, "Normal Operational Radiation Levels"

For GGNS, reactor power and nominal GGNS MS flow rate does not change as a result of the MELLLA+ operating domain expansion. The implementation of MELLLA+ may result in a greater MCO in reactor steam, with a consequential effect on selected plant radiation sources. [[ ]]. As discussed in the SAR, Section 8.4, the moisture content of the MS leaving the vessel may increase at certain times while operating in the MELLLA+ operating domain. For GGNS, an evaluation was performed using a conservatively high MCO of 0.35 wt percent. This value bounds the expected MCO as a result of operating in the MELLLA+ operating domain. This bounding increase in moisture content would increase the radiation source in the condensate demineralizers and in the FW and liquid radwaste systems by approximately 30 percent. The activity inventory in the condensate demineralizers is small compared to the RWCU demineralizers. Thus, the overall effect of the radiation source in the solid waste system will be small. The overall radiological effect of the increased moisture content is a function of the plant water radiochemistry and the levels of activated corrosion products maintained. GGNS had operated with hydrogen water chemistry (HWC) since 1998 to control stress corrosion cracking. Noble metal chemical addition was implemented in November 2010, resulting in a significant reduction in the hydrogen flow rate required to be injected into the FW and a corresponding reduction of N-16 in the steam. With the implementation of noble mechanical chemical addition, the N-16 activity concentration in the steam is bounded by the N-16 source used in the plant shielding design.

Accordingly, except as noted below, the normal operation radiation levels in the plant are expected to remain unchanged. Radiation levels are expected to increase adjacent to the condensate demineralizers, and components in the FW, liquid, and solid waste systems up to a maximum of approximately 30 percent. Because of the increase in the MCO, and due to the potential for steam leakage, the radiation levels adjacent to the Turbine Building ventilation charcoal and high efficiency particulate air (HEPA) filters may also be increased. However, there is sufficient margin to ensure that shielding is adequate. The NRC staff concludes that the existing normal operation radiation zoning will not be affected as a result of the estimated increase in radiation levels associated with MELLLA+ operation. In addition, GGNS maintains appropriate health physics for achieving doses that are as is low as reasonably achievable (ALARA). If radiation zoning during normal operation does change as a result of MELLLA+ operation, ALARA controls will help to minimize doses to plant personnel. In its September 25, 2013, letter (Reference 1), the licensee made two regulatory commitments to test and determine the MCO value at CLTP and MELLLA+ conditions. These data will be used to determine the effect of the MCO on the FAC monitoring program and for defining the MCO magnitude and trend. Based on the above, the NRC staff concludes that radiation levels will not have a significant effect on normal operational radiation levels.

<u>SAR Section 8.5.2, "Post-Shutdown Radiation Levels"</u>

For GGNS, post-operational or shutdown radiation levels depend on the decay of the fission and corrosion products in plant radioactive systems. The deposited corrosion material depends primarily on the RCS water chemistry and the cobalt impurity in the RCS. The RCS water chemistry, which is controlled by plant procedures, will remain unchanged following MELLLA+ operation. Consequently, and as discussed in the GGNS SAR, Section 8.5.1, shutdown dose rates for GGNS will reflect slight increases in areas adjacent to the condensate demineralizers, the FW, liquid and solid waste system components, and ventilation filters near the Turbine Building, but will otherwise remain unaffected by operation in the MELLLA+ domain. The NRC staff concludes there will be a minimal increase to post-shutdown radiation levels due to the increase in MCO. In addition, GGNS maintains appropriate health physics for achieving doses that are as is low as reasonably achievable (ALARA). If radiation zoning during post-shutdown operation does change as a result of MELLLA+ operation, ALARA controls will help to minimize doses to plant personnel. Based on the above, the NRC staff concludes that radiation levels will not have a significant effect on post-shutdown radiation levels.

<u>SAR Section 8.5.3, "Post-Accident Radiation Levels"</u>

Post-accident radiation levels depend primarily upon the core inventory of fission products and TS levels of radionuclides in the coolant. As power level is unchanged, operation in the MELLLA+ domain does not change the core inventory or the TS limitation on the levels of radionuclides in the coolant. Therefore, the NRC staff concludes that post-accident radiation levels are not significantly affected by operation in the MELLLA+ domain.

SAR Section 8.6.1, "Plant Gaseous Emissions"

The reactor power and the GGNS steam flow rate do not change as a result of the MELLLA+ operating domain expansion. **[[**
]]. In the MELLLA+ operating domain, MCO in the MS can increase; therefore, the licensee performed an evaluation for GGNS, using a conservatively high MCO of 0.35 wt percent. The increase in MCO results in an increase in potential iodines and particulates in airborne releases and their contribution to offsite doses by approximately 20 percent. However, doses to the public remain a small percentage of the 10 CFR 50, Appendix I, design objectives and remain within the applicable regulatory framework of 10 CFR 20. Accordingly, the NRC staff concludes that the plant gaseous emissions remain acceptable for the MELLLA+ operating domain.

SAR Section 9.0, "Reactor Safety Performance Evaluations"

Following the approved methodology in the MELLLA+ SER (Reference 30), Section 9.0 of the SAR evaluates GGNS on a plant-specific basis for the following topics.

SAR Section 9.1, "Anticipated Operational Occurrences"

SAR Section 9.1.1, "Fuel Thermal Margin Events"

Table 9-1 of the SAR presents the calculated response for the three limiting AOOs (generator load rejection without bypass, turbine trip without bypass (TTNBP), and FW controller failure (FWCF). A comparison is provided between the CLTP/80 percent flow point and CLTP/105 percent increased core flow (ICF) conditions. For the FWCF, the CLTP/105 percent ICF point is more limiting, but for turbine trip and load rejection AOOs, the 80 percent flow is more limiting. GEH attributes this effect to the CPR characteristics of GNF2 fuel. GNF2 CPR performance does not increase with flow reduction as much as GE14. This penalizes the low-flow conditions. As noted earlier, the NRC approved the applicability of GEH methods to the expanded operating domains supplement for GNF2 fuel on December 28, 2010 (Reference 29). As the SAR calculations are based on a full equilibrium core of GNF2 fuel, the NRC staff concludes the GNF2 is acceptable for use in the MELLLA+ domain. Section 3.6 of this SE and response to RAI 9 (see Appendix A of this SE) provide details of the NRC staff's review for these transients.

SAR Section 9.1.2, "Power and Flow Dependent Limits"

The NRC staff concludes that the LHGR and maximum average planar linear heat generation rate (MAPLHGR) limits are adequate for operation under MELLLA+ conditions. These limits were confirmed in the SRLR (Reference 42). This conclusion is discussed in more detail in Section 3.4.2 of this SE.

SAR Section 9.2, "Design Basis Accidents and Events of Radiological Consequences

SAR Section 9.2.1.4, "Liquid Radwaste Tank Failure"

This safety evaluation input addresses the impact of operating in the MELLLA+ domain on previously analyzed design-basis accident (DBA) radiological consequences. The criteria for which the NRC staff based its acceptance are the accident dose requirements in 10 CFR 50.67, "Accident source term." Additional criteria for which the staff based its acceptance were the accident specific design criteria provided in RG 1.183 (Reference 39).

The MELLLA+ SER indicates that a plant-specific evaluation of impact on the liquid radwaste tank failure analysis due to MELLLA+ should be performed. The liquid radwaste tank failure accident postulated by NUREG-0800 results from an unspecified event releasing the contents of the tank (or component) containing the largest inventory of radionuclides in the liquid radwaste system that is most easily transported to groundwater.

Section 9.2.1.4 of the GGNS SAR provides the plant-specific GGNS radiological consequence evaluation of the liquid radwaste tank failure accident. The GGNS's evaluation concludes that the liquid radwaste tank failure accident with expansion to the MELLLA+ operating domain meets relevant NEDO-33006P-A dispositions and regulatory requirements.

As provided in the GGNS SAR, the increase in MCO, conservatively evaluated to increase to 0.35 wt percent, is expected to increase the fission/corrosion product concentrations in the reactor steam from 0.1 wt percent to 0.35 wt percent and to increase the iodine concentrations in the reactor steam from 2 percent to 2.245 percent. The activity concentrations in the condensate system and in the related waste streams will also increase proportionally. The conservatively evaluated increase in MCO to 0.35 wt percent is also expected to increase the composite iodine concentrations in the equipment drain collector tank by 3 percent. Thus, MELLLA+ operation is expected to result in a 3 percent increase in the dose consequences. As shown in Table 9-4 of the LAR (Reference 1), the MELLLA+ accident doses for the liquid radwaste tank failure were determined to be within the applicable regulatory limits. The radiological consequence due to the liquid release pathway of a liquid radwaste tank failure is addressed in the GGNS UFSAR, Section 15.7.3. The limiting event for this pathway is the RWCU system phase separator decay tank. The radionuclide inventory in the radwaste tanks, adjusted for higher MCO, is bounded by the inventory used in the liquid radwaste tank failure analysis currently presented in the UFSAR. Therefore, the dose calculation described in the UFSAR for the liquid release pathway of a liquid radwaste tank failure remains bounding. The NRC staff agrees that GGNS meets all NEDO-33006P-A dispositions for the liquid radwaste tank failure and is bounded by the inventory used in the liquid radwaste tank failure analysis currently presented in the UFSAR, which meets regulatory requirements.

The NRC staff reviewed the dose consequences of the licensee's proposed changes. Since there are no major modifications to plant equipment, and no increases in the design basis operating pressure, power, core inventory source terms, steam flow rate, and FW flow rate, the staff finds that the GGNS's DBA dose consequence evaluation is reasonable. Furthermore, all dose consequences relating to the proposed expansion of the power-to-flow map to MELLLA+ are bounded by the currently licensed DBAs.

Based on the above, the NRC staff concludes that the GGNS MELLLA+ operation is bounded by the existing analyses in the approved NEDO-33006P-A (Reference 56) and the GGNS UFSAR. The radiological dose consequences for all accidents remain below the design criteria specified in 10 CFR 50.67 and the accident specific design criteria outlined in RG 1.183 (Reference 39). The NRC staff concludes that the implementation of MELLLA+ at GGNS is acceptable with regard to radiological consequences. In addition, Entergy has made two regulatory commitments to test and determine the MCO value at CLTP and MELLLA+ conditions. These data will be used to determine the effect of the MCO on the FAC monitoring program and for defining the MCO magnitude and trend.

SAR Section 9.3, "Special Events"

Special events (ATWS events), except for SBO, are discussed in more detail in Section 3.7 of this SE.

SAR Section 10.4, "Testing"

SAR Section 10.4.1, "Steam Separator-Dryer Performance"; SAR Section 10.4.2, "Average Power Range Monitor Calibration"; SAR Section 10.4.3, "Core Performance"; SAR Section 10.4.4, "Pressure Regulator"; SAR Section 10.4.5, "Water Level Setpoint Changes"; and SAR Section 10.4.6, "Neutron Flux Noise Surveillance"

The NRC staff concludes that the proposed testing above are reasonable and acceptable for the approval for the operation in the MELLLA+ domain. Several tests were performed during the EPU startup.

SAR Section 10.5, "Individual Plant-Examination"; SAR Section 10.5.1, "Initiating Event Categories and Frequency"; SAR Section 10.5.2, "Component and System Reliability"; SAR Section 10.5.3, "Operator Response"; SAR Section 10.5.4, "Success Criteria"; SAR Section 10.5.5, "External Events"; SAR Section 10.5.6, "Shutdown Risks"; and SAR Section 10.5.7, "PRA Quality"

The NRC staff reviewed the GGNS's LAR and determined that it was not risk-informed[1], but the GGNS's SAR did include, in Attachment 5, risk insights related to the implementation of the MELLLA+ domain. Specifically, the licensee augmented the generic risk discussion contained in MELLLA+ LTR with plant-specific information on initiating event frequencies, component reliability, operator response, success criteria, external events, shutdown risk, and probabilistic risk assessment (PRA) quality. The licensee reported an increase in core damage frequency (CDF) of 2×10^{-8} / year and an increase in large early release frequency (LERF) of 1×10^{-9} /year primarily due to slight changes to human error probabilities associated with ATWS sequences.

[1] Review Standard-001 defines a "risk-informed" LAR as one that requests relaxation of deterministic requirements based in part on risk arguments.

Consistent with the NRC's guidance on non-risk-informed LARs (SRP, Chapter 19.2, Appendix D, Reference 35(s)), the NRC staff reviewed Attachment 5 to the GGNS SAR to determine whether "special circumstances" were present (e.g., a risk increase exceeding the RG 1.174 acceptance guidelines (Reference 38)) that would warrant a more detailed risk evaluation. Based on the risk information provided by the licensee, the NRC staff concludes that the expected increase in risk associated with implementation of MELLLA+ at GGNS would be well within the risk acceptance guidelines delineated by RG 1.174. Therefore, the NRC staff's review did not identify any "special circumstances" that would warrant an in-depth PRA review.

SAR Section 10.6, "Operator Training and Human Factors"

In accordance with the generic risk categories established in Appendix A to NUREG-1764 (Reference 41), the tasks under review are involved in the safety injection sequence actions involving risk-important systems are, therefore, considered "risk-important." Due to this estimated risk importance, the NRC staff performed a "Level One" review, the most stringent of the graded reviews possible under the guidance of NUREG-1764.

Technical Evaluation

Description of Operator Actions Added/Changed/Deleted

There are no new operator actions that are added as a result of this amendment. The GGNS SAR, Section 10.5.3, indicates that the operator actions are, "basically the same as for extended power uprate (EPU)." However, three operator actions that have increased importance are identified that are necessary for mitigation of ATWS and ATWS with instability events.

Although there are no new operator actions, there may be changes to operator action timing for ATWS level/power control, and there is potential for SRV cycling. As such, the licensee may change some guided actions in the emergency operating procedures (EOPs) to immediate operator actions. If used, operators will be specially trained for these actions and they will be memorized. This is expected to improve operator performance on these tasks and improve operator response time, thereby increasing time margin. In the letter dated February 19, 2015 (Reference 20), the licensee has proposed a license condition to inform the NRC if it decides to make this change to the EOPs.

In addition, the licensee has proposed a license condition to validate all three time-critical operator actions. This includes a commitment to perform the most time-sensitive operator action in 90 seconds, instead of the 120 seconds available (as indicated by the accident analysis) during verification and validation (V&V). This is more conservative than the previously approved action times for EPU operations (Reference 24); therefore, the NRC staff concludes that this decrease in operator time for the ATWS-I event to 90 seconds is acceptable.

Operating Experience Review

The licensee described how an operating experience review (OER) was conducted in the September 10, 2014, RAI response letter (Reference 9). The licensee searched the World Association of Nuclear Operators and Institute of Nuclear Power Operations databases for operating experience related to operation in both the MELLLA and MELLLA+ domains. However, there was no applicable data pertaining to MELLLA+ operations. The licensee also solicited operating experience information from the Monticello plant, which is currently licensed for MELLLA+ operation.

The licensee is relying on existing human-system interfaces (HSI) with minimal modifications to perform safety-related functions. The only changes are to a small number of setpoints and alarms, and minor changes to the operating power/core flow map. Section 3.2 of NUREG-1764 (Reference 41) indicates that OER is not necessary when there are no changes to HSI. The types of changes described are minimal and are unlikely to have significant effects on performance, as long as operators are trained appropriately and have adequate procedures. The licensee performed an OER, and therefore, has met or exceeded what is required of this criterion. Therefore, the NRC staff finds the licensee's treatment of this review element is acceptable.

Functional Requirements Analysis and Function Allocation

The GGNS's SAR, Section 10.5.3, indicates that MELLLA+ operation does not institute any changes to the operator actions approved for EPU conditions. The proposed changes do not affect the plant's functional requirements analysis or the function allocation. Criterion 3.3 of NUREG-1764 indicates that if no changes to the functional requirements or function allocation occur, then this review element is not necessary.

Task Analysis

Operation within the MELLLA+ domain does not require significant changes to operator actions. Only limited changes to the HSI are necessary, such as the graphics for the operating power/core flow map and a limited number of setpoints.

By letter dated September 11, 2014 (Reference 9), the RAI response to question 1 indicates that operator actions remain the same as the actions for the previously approved EPU, and there is no increase in operator workload. The response to question 3 indicates that there are no changes necessary to existing task analyses as a result of MELLLA+ operations.

The actions associated with this proposed change are the same as previously approved actions under EPU conditions. The actions are proceduralized and the existing training program reduces the likelihood that operators will make errors.

No changes are needed for the existing task analysis. There are no changes to the operator actions, and changes to the interfaces are minimal and unlikely to affect performance. Therefore, the NRC staff finds the licensee's treatment of this review element is acceptable.

Staffing

Based on the lack of changes to current operations, no new or additional staff are required, nor are there any new or additional qualifications required to perform the actions within the time constraints established. Operation in the MELLLA+ domain is not expected to increase operator workload. The NRC staff concludes that no additional staffing, qualifications, or changes thereto are required; therefore, the NRC staff concludes that the licensee's treatment of this review element is acceptable.

Probabilistic Risk and Human Reliability Analyses

The NRC staff reviewed the GGNS LAR and determined that it was not risk-informed[2] but did include risk insights related to the implementation of MELLLA+. As noted above, the NRC staff concludes that although there was a small increase in CDF, there were no special circumstances that warrant more detailed risk evaluations.

The licensee uses a structured, systematic, and auditable risk assessment process, which has been used to update the CDF and LERF. Changes to the associated human error probabilities cause only a minimal increase to risk and do not necessitate changes to operator actions. These human actions are consistent with the existing approved ATWS operator actions. Therefore, the NRC staff concludes that the licensee's treatment of this review element is acceptable.

Human-System Interface Design

The GGNS SAR, Section 10.6, "Operator Training and Human Factors," describes minimal changes to HSIs. Changes include an update to the computer display of the power/flow map and changes to some instrument and alarm setpoints. This update includes adding an additional region to the existing core/flow map. Hardware changes to the video display unit that displays the core/flow map are not necessary.

There are no planned updates of controls, displays, or alarms from analog to digital systems. No changes to instrument or alarm hardware are necessary. SAR, Section 5.3, describes how changes to TS AVs and NTSP changes will be conducted in accordance with previously approved change control procedures.

The DSS-CD was installed with previous plant modifications. The current GGNS LAR, if approved, will allow the licensee to begin using this function imbedded in the existing system.

2 Review Standard-001 defines a "risk-informed" LAR as one that requests relaxation of deterministic requirements based in part on risk arguments. Chapter 19, Appendix D of the Standard Review Plan contains guidance for when risk insights may impact a non-risk-informed LAR.

The change involves removing a series of jumpers from inside the hardware and does not affect the existing HSI.

The proposed change does not alter the HSI in any significant way. The licensee uses training to ensure that operators are aware of the minor changes to the CF map and instrument/alarm setpoint changes. Therefore, the NRC staff concludes that the licensee's treatment of this review element is acceptable.

<u>Procedure Design</u>

The GGNS SAR, Section 10.9, "Emergency and Abnormal Operating Procedures," addresses changes to EOPs and Abnormal Operating Procedures (AOP) as a result of operation in the MELLLA+ domain. No changes are expected to EOPs or AOPs. Both EOPs and AOPs will be reviewed and revised using existing plant protocols. Any changes identified will be included in operator training (SAR, Sections 10.9.1 and 10.9.2). The licensee has also made regulatory commitments with regard to updating the AOPS and EOPs as needed. This commitment will be completed within 90 days of the approval of the MELLLA+ amendment.

The licensee is considering changing some steps of the EOPs by making them immediate operator actions. This will have the benefit of increasing the speed of operator reaction and may potentially increase the time margin for some time-critical operator actions. This decision will not be finalized until the V&V process is complete. In a letter dated February 19, 2015 (Reference 20), the licensee proposed a license condition to notify the NRC if steps of the EOP will be made into immediate operator actions.

The SAR, Section 5.1, describes the use of normal plant procedures and surveillance procedures for the adjustment of APRMs, SRMs, and IRMs.

The licensee uses existing procedure change processes to control changes to procedures. The docketed information submitted by the licensee does not require changes to procedures. The use of the regulatory commitment assures that any changes that may be necessary to EOPs regarding the use of immediate actions will be validated, and the NRC staff will be informed. Therefore, the NRC staff finds the licensee's treatment of this review element is acceptable.

<u>Training Program Design</u>

The GGNS SAR, Section 10.6, describes how "just-in-time" classroom training will include the important considerations with respect to operation in the MELLLA+ domain, changes to procedures, setpoints, and HSIs. Classroom training may be supplemented with simulator training. The licensee indicates that enhanced simulator training for ATWS events will occur. The existing approved training program will be used to select specific topics to be included. Training will include any changes to TSs, EOPs, and plant systems. Changes to the simulator will be evaluated using applicable American National Standards Institute (ANSI) standards.

The training program, as described, is consistent with the current licensing basis and uses approved methods to incorporate any changes to the HSI, procedures, and operational considerations into the training program. The proposed license condition commitment ensures that training is complete prior to MELLLA+ operation. Therefore, the NRC staff finds that the licensee's treatment of this review element is acceptable.

Human Factors Verification & Validation

The GGNS SAR, Section 10.6, indicates that simulator validation will occur in accordance with the applicable ANSI standards.

In the February 19, 2015, submittal, the licensee provided a license condition to complete operator training of all crews and to conduct human factors V&V activities prior to beginning MELLLA+ operation. This license condition includes validating all time-critical operator actions including the most time-constrained operator action in less than 90 seconds (30 seconds sooner than what is deemed necessary by the licensee analysis). This additional margin provides additional assurance that the operators are capable of completing actions in the time available. The license condition also includes a requirement that the licensee report the results of the validation to the NRC within 60 days of completion of the validation testing.

The license condition provides reasonable assurance that operators will be capable of meeting or exceeding applicable operator action times. Therefore, the NRC staff concludes that the licensee's treatment of this review element is acceptable.

Human Performance Monitoring Strategy

There are no changes being made to operator actions as a result of MELLLA+ operation. Since the system provides automatic functions for the RPS, which are the same as the existing analog systems, there are no changes in required operator actions. Therefore, there is no need to amend the formal human performance program. The GGNS SAR, Section 10.6, indicates that the licensee will store data from operations in the MELLLA+ domain for incorporation into future training. The NRC staff concludes that the licensee's treatment of this review element is acceptable. The existing program is adequate to monitor human performance under these conditions.

Conclusion

Based on the information currently provided in the LAR, the associated RAI responses, and the implementation of the associated license condition, the NRC staff concludes that the proposed changes in support of GGNS MELLLA+ operation are acceptable. Approval of this LAR is based upon appropriate human factors V&V implementation, adequate results, and the submission of docketed information regarding the use of immediate operator actions, or the confirmation that they will not be used, as described in the License Condition 49. This license condition is also discussed in Section 4.2 of this SE.

SAR Section 10.9, "Emergency and Abnormal Operating Procedures"

EOPs and AOPs can be affected by operating in the MELLLA+ domain.

SAR Section 10.9.1, "Emergency Operating Procedures"

The EOPs include variables and limit curves, which define conditions where operator actions are indicated. The EOPs remain symptom-based, and thus the operator actions remain unchanged. Therefore, the MELLLA+ operating domain expansion is not expected to affect the GGNS EOPs. In accordance with MELLLA+ LTR SER, Limitation and Condition 12.23.4 (Reference 30), the EOPs will be reviewed for any effect and revised as necessary prior to implementation of the MELLLA+ operating domain expansion. Any changes identified to the EOPs will be included in the operator training to be conducted prior to implementation of MELLLA+. In its letter dated September 25, 2013 (Reference 1), Entergy has made a regulatory commitment to review and revise its EOPs as necessary as result of operating in the MELLLA+ domain. This commitment will be completed within 90 days of the approval of the MELLLA+ amendment.

SAR Section 10.9.2, "Abnormal Operating Procedures"

AOPs include event-based operator actions. No significant AOP revisions are expected as a result of the MELLLA+ operating domain expansion. However, the AOPs will be reviewed for any effect and revised as necessary prior to implementation of the MELLLA+ operating domain expansion. Any changes identified to the AOPs will be included in the operator training to be conducted prior to implementation of MELLLA+. Entergy has made a regulatory commitment with regard to the above. This commitment will be completed within 90 days of the approval of the MELLLA+ amendment.

3.4. NRC Evaluation of SAR Section 2.0, "Reactor Core and Fuel Performance"

As noted above, even though the licensee concludes that the topics in this section met the generic disposition of the MELLLA+ LTR, the NRC staff performed a review to confirm these conclusions. The NRC staff used RS-001 (Reference 34) as a reference in conducting the MELLLA+ review. Although MELLLA+ is not a power uprate, and RS-001 guidance is not wholly applicable, RS-001 provides a reasonable framework for review of this application. The NRC staff recognizes that there are sections in RS-001 that are unnecessary for the MELLLA+ application review. RS-001 specifies the following review areas that were evaluated for the proposed extension of the operating domain:

1) Fuel System Design (Reviewed in SE Section 3.4)
2) Nuclear Design (Reviewed in SE Section 3.4)
3) Thermal and Hydraulic Design (Reviewed in SE Section 3.4)
4) Emergency Systems (Reviewed in SE Section 3.5)
5) Accident and Transient Analyses (Reviewed in SE Sections 3.6 and 3.7)

The staff from Oak Ridge National Laboratory (ORNL) assisted the NRC staff in its review.

3.4.1 Fuel System Design

Regulatory Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, channel boxes, and reactivity control rods. The NRC staff reviewed the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and AOOs, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NRC staff review covered fuel system damage mechanisms, limiting values for important parameters, performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46 insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) GDC 10 insofar as it requires that the reactor core be designed with the appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (3) GDC 27 insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained; and (4) GDC 35 insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA. Specific review criteria are contained in the SRP, Section 4.2 (Reference 35(3)) and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The NRC staff reviewed the impact on the fuel system of the proposed MELLLA+ operating domain extension based on the licensee-provided analyses for normal operation, AOOs, and infrequent and special events. The complete staff evaluation of these results is documented in Sections 3.6 and 3.7 of this SE. As seen in that evaluation, operation at the lower MELLLA+ flows has an impact on transient response, and the effect on fuel becomes slightly more severe for some events. To mitigate these events, the licensee proposes to use more restrictive setpoints consistent with the AOO ΔCPR methodology so that the final SLMCPR is maintained constant. The licensee analyses demonstrate that with the proposed GGNS MELLLA+ setpoints, fuel damage is not expected for any AOO or any unanalyzed infrequent incident or special events, and core coolability is always maintained.

Conclusion

The NRC staff reviewed the licensee's analyses related to the effects of the proposed operating domain extension on the fuel system design of the fuel assemblies, control systems, and reactor core. The staff concludes that the licensee has adequately accounted for the effects of the proposed operating domain extension on the fuel system and demonstrated that (1) the fuel system will not likely be damaged as a result of normal operation and AOOs, (2) the fuel system damage, should it happen, is not likely to be so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures has not been underestimated for postulated accidents, and (4) coolability is likely to be maintained. Thus, the staff concludes that the impact on fuel while operating with the more restrictive setpoints at the lower MELLLA+ flows is minimal. Based on the above, the staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, GDC 10, GDC 27, and GDC 35, following implementation of the proposed operating domain extension. Therefore, the NRC staff finds the proposed operating domain extension acceptable with respect to the fuel system design.

3.4.2 Nuclear Design

Regulatory Evaluation

The NRC staff reviewed the nuclear design of the fuel assemblies, control systems, and reactor core, to ensure that fuel design limits will not be exceeded during normal operation and anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. The staff's review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns, reactivity worths, criticality, burnup, and vessel irradiation.

The NRC's acceptance criteria are based on (1) GDC 10 insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC 11 insofar as it requires that the reactor core be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity; (3) GDC 12 insofar as it requires that the reactor core be designed to assure that power oscillations which can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed; (4) GDC 13 insofar as it requires that instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs, and accident conditions, and to maintain the variables and systems within prescribed operating ranges; (5) GDC 20 insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to assure that specified acceptable fuel design limits are not exceeded as a result of AOOs and to sense accident conditions and to initiate operation of systems and components important to safety; (6) GDC 25 insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (7) GDC 26 insofar as it requires that two independent reactivity control systems of different design be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal operation; (8) GDC 27 insofar as it requires that insofar as it requires that the reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained; and (9) GDC 28 insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core. Specific review criteria are contained in SRP Section 4.3 (Reference 35(f)) and other guidance provided in Matrix 8 of RS-001 (Reference 34).

Technical Evaluation

This evaluation addresses Operating Limits, Monitoring and Control, and Reactivity Control.

Operating Limits

For this topic specifically, the NRC staff acceptance criteria was based on the following GDC in 10 CFR 50, Appendix A.

GDC 10 specifies the requirements for core operating limits. GDC 10 is met by operating the plant within established operating limits. The OLMCPR and the MAPLHGR limits are designed to protect the fuel during normal operation, as well as during anticipated transients, from exceeding SAFDLs.

The NRC staff reviewed the design changes between the GGNS EPU core design and a reference MELLLA+ core design in terms of its impact on compliance with GDC 10. The NRC staff notes that the core and fuel design remain unchanged, and a full load of GNF2 is used.

The SLMCPR is calculated based on the actual core loading pattern for each reload core, and the results are reported in the SRLR. In the event that the cycle-specific SLMCPR is not bounded by the current GGNS TS value, GGNS must implement a license amendment to change the TS. As required by the MELLLA+ SER (Reference 30), the SLMCPR is calculated at different operating conditions for every reload core. The specified conditions for GGNS include 100 percent CLTP/100 percent flow, 100 percent CLTP/80 percent flow, 100 percent CLTP/105 percent flow, and 80.6 percent CLTP/55 percent flow. The calculated SLMCPR values include the adders required by the methods SER (Reference 31) for operation in the MELLLA+ domain (see response to RAI 2 in letter dated October 2, 2014 (Reference 10), for more details).

For the current Cycle 20 with GNF2 fuel, the SLMCPR is 1.11 for TLO and 1.14 for SLO. For MELLLA+ operation, the licensee will increase the SLMCPR to 1.15 as indicated in the SRLR for TLO and SLO. The two-loop SLMCPR value is increased 0.04, primarily due to condition 12.6 in the MELLLA+ SER, which requires the application of the SLO core flow uncertainty to the TLO calculation. The licensee submitted a license amendment request on November 21, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14325A520), to increase the SLMCPR from 1.11 to 1.15 for TLO and from 1.14 to 1.15 for SLO. By letter dated August 18, 2015 (ADAMS Accession No. ML15229A213), the NRC staff issued Amendment No. 203 approving the new SLMCPR values needed to support operation in the MELLLA+ domain.

The OLMCPR is calculated by adding the change in MCPR due to the limiting AOO event to the SLMCPR. The OLMCPR for GGNS is determined on a cycle-specific basis from the results of the reload transient analysis, which is documented in the SRLR. A 0.01 adder is applied to the resulting OLMCPR as required by the methods SER (Reference 31). The final value of the OLMCPR is documented in the COLR. Based on the generic results documented in the MELLLA+ SER (Reference 30), and the reference transient analyses documented in the GGNS SAR, Section 9 (Reference 28), the OLMCPR is only expected to change by ~0.016 for MELLLA+ operation because of the reduced CPR performance of GNF2 fuel at low flows (see Section 3.1.5 of this SER).

The LHGR and MAPLHGR operating limits are calculated for each reload fuel bundle design. The power and flow dependent LHGR multipliers are sufficient to provide adequate protection for the off-rated condition from an ECCS-LOCA analysis perspective. The limits are documented in the cycle-specific COLR.

Section 4.3 of the GGNS SAR presents results for a LOCA analysis at different initial conditions. The licensee performed LOCA analyses for large break and small breaks at different power/flow conditions. Calculations were performed at top-peaked and mid-peak power shapes. The ECCS-LOCA analyses was performed for all state points in the upper boundary of the expanded operating domain, including the minimum core flow state points, the transition state point, and the 55 W_T state point. The limiting 10 CFR 50, Appendix K, large break single failure for GGNS MELLLA+ operating domain remains the RSLB with the HPCS diesel generator failure. The GGNS PCT, local cladding oxidation, and core-wide metal water reaction results were calculated with SAFER using the PRIME T-M performance methodology. The upper bound PCT is bounded by the licensing PCT, which is 1730 °F. This is below the 2200 °F, and therefore, satisfies 10 CFR 50.46 acceptance criterion.

Monitoring and Control

For this topic specifically, the NRC staff acceptance criteria was based on the GDC 13 in 10 CFR 50, Appendix A, that specifies the requirements for instrumentation to monitor variables affecting the fission process. Maneuvering within the MELLLA+ operating domain is performed by either controlling the recirculation flow or moving control rods. GDC 13 requires that instrumentation be provided to ensure that the operation is within prescribed operating ranges.

The design changes to incorporate MELLLA+ do not include any changes to the neutron monitoring system (NMS) or the flow instrumentation. Nevertheless, the NRC staff reviewed the effects of operation in the expanded domain on instrumentation performance, and thus, the adequacy of the NMS to meet the requirements of GDC 13. At the MELLLA+ low corner (Point C of Figure 1-1 of the SAR), the power-to-flow ratio is maximized, and there is the potential to encounter void formation in the bypass region. In RAI-03 by letter dated August 26, 2014 (Reference 7), the staff requested that the licensee provide an assessment of bypass void formation at point C. Furthermore, the staff requested that the licensee determine the effects of bypass void formation on LPRM. The evaluation was performed, and the bypass void fraction is expected to be less than 4.7 percent at the highest TIP elevation and less than 2.4 percent at the LPRM D level elevation (see Table 5-1 of the SAR). This value is calculated using the ISCOR hot channel methodology, which is conservative, because it neglects cross flow between bundles in the bypass region; thus, the bypass voids are expected to be lower than the 4.7 percent value calculated. Proper LPRM calibration requires that the bypass void be maintained to a value less than 5 percent. Therefore, the staff concludes that bypass voiding under MELLLA+ conditions in GGNS does not affect the LPRM or TIP instrumentation adversely and that the GGNS instrumentation and control systems are adequate to fulfill the requirements of GDC 13 under MELLLA+ operating conditions.

Reactivity Control

For this topic specifically, the NRC staff acceptance criteria was based on the following GDC in 10 CFR 50, Appendix A.

GDC 20, 25, 26, 27, and 28 specify the requirements for the reactivity control systems. Power control is achieved in the expanded operating domain by controlling core reactivity with control blades, as well as recirculation flow.

GDC 20 and 25 are met by the RPS and the scram function of the control rod system. These systems are unaffected by the implementation of the MELLLA+ domain.

GDC 26 and 27 are met by the control rod system and the SLCS. The CRD system is unaffected by the implementation of the MELLLA+ domain. The SLCS TS relief valve setpoint value was increased from 1340 psig to 1370 psig and discussed in Section 3.5.5 of this SE.

As discussed in Section 3.2.8, SAR Section 9.2.1.1, the licensee stated that operation in the MELLLA+ operating domain does not affect the CRDA source terms, and assumed peaking factor remains bounding. In addition, there are no changes to the removal, transport, or dose conversion assumptions for this event. As such radiological consequences and barrier integrity during postulated CRDAs remain unchanged and GGNS continues to comply with GDC 28. In addition, the most limiting conditions occur during low power operation and are, therefore, unaffected by the MELLLA+ implementation which is only at higher power levels.

Conclusion

The NRC staff reviewed the licensee's analyses related to the effect of the proposed operating domain extension on the nuclear design of the fuel assemblies, control systems, and reactor core. The staff concludes that the licensee has adequately accounted for the effects of the proposed operating domain extension on the nuclear design. The licensee has further demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation, and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28. Therefore, the NRC staff finds the proposed operating domain extension acceptable with respect to the nuclear design.

3.4.3 Thermal and Hydraulic Design

Regulatory Evaluation

The NRC staff reviewed the thermal and hydraulic design of the core and the RCS to confirm that the design (1) has been accomplished using acceptable analytical methods; (2) is equivalent to, or a justified extrapolation, from proven designs; (3) provides acceptable margins of safety from conditions, which would lead to fuel damage during normal reactor operation and AOOs; and (4) is not susceptible to thermal-hydraulic instability. The review also covered (1) hydraulic loads on the core and RCS components during normal operation and DBA conditions and (2) core thermal-hydraulic stability under normal operation and ATWS events. The NRC's acceptance criteria are based on (1) GDC 10 insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs, and (2) GDC 12 insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can reliably and readily be detected and suppressed. Specific review criteria are contained in SRP Section 4.4 Reference 35(g)) and other guidance provided in Matrix 8 of RS-001 (Reference 34).

Technical Evaluation

Analytical Methods

The NRC staff reviewed the analytical methods utilized by the licensee. A comprehensive list of the codes used to support the licensee's analyses is documented in the GGNS SAR, Table 1-1 (Reference 28). As described in the footnotes to Table 1-1, not all codes used have an explicit staff SER associated with them; however, sufficient regulatory bases are provided for their use.

The following exceptions are noted:

1. The ISCOR code does not have an explicitly approved SER; however, the approval SER for NEDE-24011P, Revision 0 (Reference 57) digital computer code referred to in NEDE-24011P, Revision 0, SER is indeed ISCOR.

2. A similar situation occurs with the STEMP code, which is used to calculate the SP temperature using basic energy conservation equations. STEMP was referenced in the approval of NEDE-24222 (Reference 58).

3. The LAMB code is explicitly approved for use in ECCS-LOCA applications, but it is not explicitly approved for use in RIPDs and containment response. However, this is simply an extension of the approved use, and the models used are those of the approved ECCS-LOCA application.

4. TRACG04 is currently approved for use in DSS-CD and ATWS analyses and has been used for ATWS best estimate calculations; however, the licensing basis ATWS analyses are based on ODYN.

5. Following approval of Amendment 26 of GESTAR II (Reference 59), GEH implemented TGBLA06 and PANAC11.

In the SAR, Table 1-1, "Computer Codes Used in the MELLLA+ SAR Evaluations," the licensee has provided the approval status for the codes listed above and when, and by which, GEH LTRS they were approved. Thus, the NRC staff concludes that all the methods used in the SAR are either approved or are an acceptable extension of an approved code.

Equivalency to Proven Designs

The proposed MELLLA+ operating domain is similar in design to the power-flow operating domain currently in use by GGNS. The primary difference is the higher power-to-flow ratio in

the MELLLA+ corner, which results in higher operating void fraction and higher operating power when the recirculation pumps are tripped that affect ATWS performance.

Steady State Operation

Table-1 shows a summary of the GGNS steady state operating conditions (extracted from Tables 1-2 and 1-3 of the GGNS SAR).

Table 1. GGNS Operating Conditions

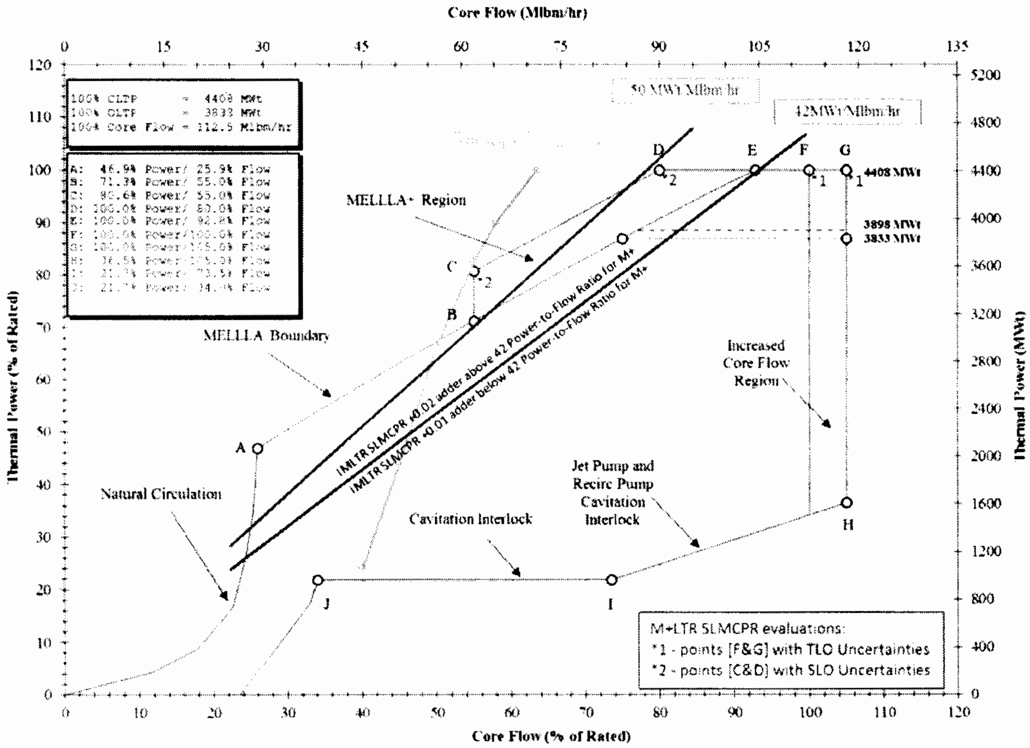
Parameter	MELLLA 100% CLTP, 92.8% CF	MELLLA+ 100% CLTP, 80% CF	MELLLA+ 80.6% CLTP, 55% CF
Thermal Power (MWt)	4408	4408	3553
Dome Pressure pounds per square inch absolute (psia)	1040	1040	1013
Steam Flow Rate Million Pounds Per Hour (Mlb/hr)	18.967	18.964	14.796
FW Flow Rate (Mlb/hr)	18.934	18.931	14.763
FW Temperature (°F)	420.0	420.0	397.0
CF (Mlb/hr)	104.4	90.0	61.9
Core Inlet Enthalpy British Thermal Unit/Pounds Mass (BTU/lbm)	523.1	518.7	505.2
Core Pressure Drop (psi)	20.0	15.3	6.8
Core Average Void Fraction	0.52	0.55	0.55
Average Core Exit Void Fraction	0.73	0.76	0.77

Table 2. GGNS and Gulf Power-to-Flow Ratios

Operating Domain	Point on Power/Flow Map	Core Thermal Power (MWt /%CLTP)	CF (Mlbm/hr/% Rated)	Power-to-Flow Ratio (MWt / Mlbm/hr)
Current Operating Domain 92.8% Rated Core Flow (RCF)	E	4,408 / 100.0	104.4 / 92.8	42.2
MELLLA+ Operating Domain 80% RCF	D	4,408 / 100.0	90.0 / 80.0	49.0
MELLLA+ Operating Domain 55% RCF	C	3,553 / 80.6	61.9 / 55.0	57.4

As seen in Table 2, the power density at Point C (55 percent flow, see Figure 1 of this SE) is 57.4 MWt/Mlbm/hr, which is greater than the action threshold of 50 MWt/Mlbm/hr set by Limitation 9.3 of the Methods LTR and associated SER (References 7 and 10, respectively). During the evaluation of the methods LTR, the NRC staff reviewed power distribution uncertainties up to power-to-flow ratios of 42 MWt/Mlbm/hr, and found that an uncertainty of 0.02 should be added to the SLMCPR to cover operation above 42 MWt/Mlbm/hr (see Figure 2). The reason for this limitation is that insufficient data were available to judge power distribution uncertainties at the higher void fraction levels. Extremely high void fractions result in increased errors in cross section generation and challenge some of the assumptions in modern nodal neutronic methods because of the harder neutron spectrum.

Figure 2. Illustration of Power-to-Flow Ratio Requirements from Limitation 9.3



Grand Gulf EPU/MELLLA+

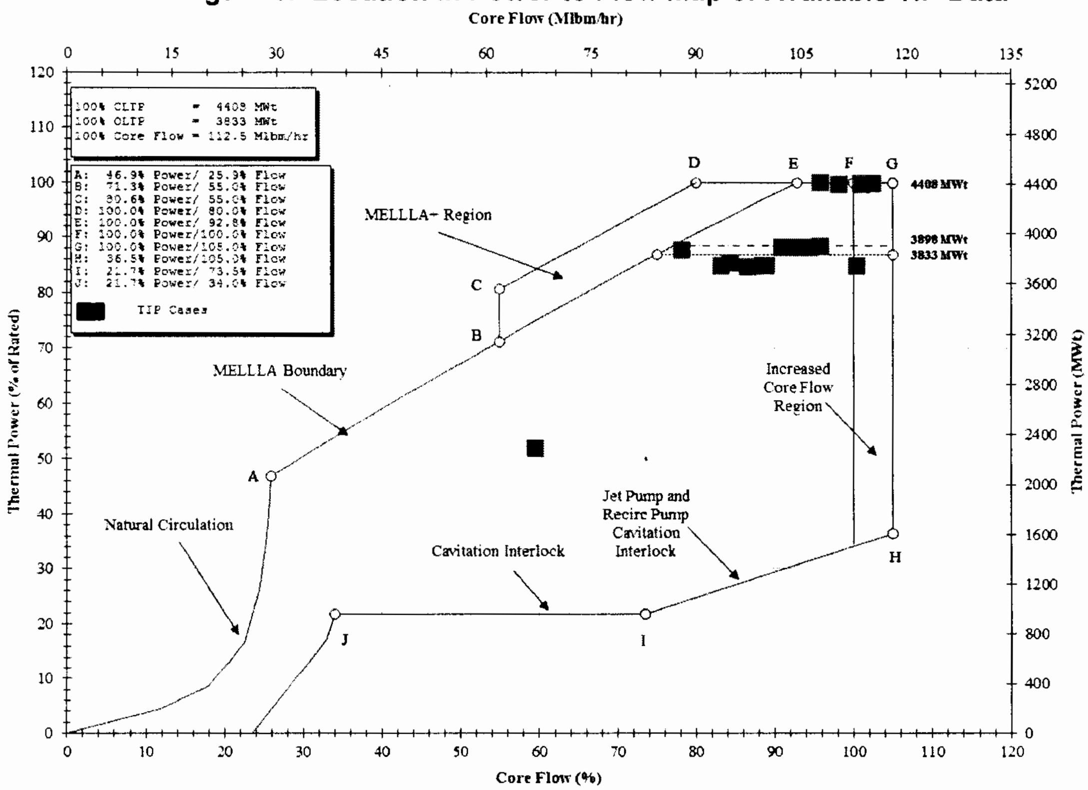
For operation at power-flow greater than 50 MWt/Mlbm/hr (for example, at point C in Figure 2, above), the NRC staff SER required a case-specific evaluation to ensure that the particular plant is not an outlier and has unusual uncertainty values. The NRC staff also required an additional penalty on the SLMCPR by using SLO uncertainties, even though SLO operation is not allowed under MELLLA+. This restriction applies to Point C and D in Figure 2. For GGNS Cycle 20 (the first MELLLA+ implementation), use of the SLO flow uncertainties is estimated to increase the TLO SLMCPR from 1.11 to 1.15, equivalent to a 0.04 SLMCPR penalty, which is quite significant.

In addition to the 42 and 50 power-to-flow lines (red lines), Figure 2 shows the line in the power-to -flow map where the TIP exit void fraction in the bypass region is 5 percent (orange line). Bypass void fractions can affect the calibration of LPRM or TIP detectors because they are calibrated at full power where there is no bypass voiding. The MELLLA+ SER (Reference 30) requires that the bypass void fraction be lower than 5 percent to prevent the decalibration issues. The calculation shown in Figure 2 confirms that bypass voiding is not a problem in GGNS operating in the MELLLA+ domain.

Following the guidelines from Limitation 9.3 (Reference 30), the NRC staff reviewed, on a plant-specific basis, the power distribution uncertainties for GGNS. This review was based on a comparison of TIP data provided by the licensee against PANACEA calculated power distributions.

Figure 3 shows the locations in the power-to-flow map where TIP measurements were performed in GGNS for the last three cycles (18-20). Measurements have been taken in GGNS with power densities as high as 44.07 MWt/Mlbm/hr. Analyses of these data show that the power distribution root mean squared error is <1 percent, with the error increasing towards end of cycle, but < 1.5 percent for all points.

Figure 3. Location in Power to Flow Map of Available TIP Data



Grand Gulf EPU/MELLLA+ TIP Cases

Transient Response

The licensee has provided analyses for normal operation, AOOs, and special events. The complete NRC staff evaluation of these results is documented in Section 3.6 of this SE. As seen in that evaluation, operation at lower flows in the MELLLA+ domain has an impact on transient response. Calculations (see Table 9-1 of the GGNS SAR (Reference 28)) show that the limiting

AOO is the load rejection with no bypass (LRNBP). For this case, the AOO ΔCPR is 0.252 when initiating from the 80 percent flow condition and 0.236 from the 105 percent flow condition. This results in an increase of OLMCPR.

The OLMCPR steady state limits are calculated on a cycle-specific basis to maintain the same margin to the SLMCPR during transients. The transient ΔCPR, which defines the OLMCPR, is calculated for all transients affected by the MELLLA+ extension. In this way, the limiting transient event initiating from inside the MELLLA+ region has the same margin to the SLMCPR than before the MELLLA+ domain was implemented.

Stability

As required by the MELLLA+ LTR (Reference 30), GGNS will implement the DSS-CD solution consistent with the limitations and conditions in the applicable DSS-CD SER. DSS-CD SER (Reference 32) specifies the procedures needed to implement a new fuel transition. These procedures were followed by GGNS, and the results are presented in SAR Section 2.4. In addition the NRC staff notes that in accordance with DSS-CD LTR SER Limitation and Condition 5.1, because GGNS is implementing DSS-CD using the NRC approved GEH Option III platform, a plant-specific review is not required. There were no changes proposed in the bounding uncertainty or in the process to bound the uncertainty in the MCPR. Based on the above, the NRC staff concludes these results are acceptable for transitioning from Option III to DSS-CD with GNF2 fuel. GGNS will be the first implementation of GNF2 fuel for a transition from Option III to DSS-CD.

In order to use a higher DSS-CD setpoint value **[[**

]]. The calculations documented in Table 2-6 of the GGNS SAR confirm that **[[**

]], because they follow the approved procedure established in the DSS-CD SER, and the calculated final minimum critical power ratio (FMCPR) is greater than the SLMCPR for the representative cases studied.

[[

]], based on the methodology described in the GGNS SAR, Section 2.4). **[[**

]] the process described in Section 2.4 of the SAR and in Section 6 of the DSS-CD LTR (Reference 32).

The oscillation power range monitor (OPRM) performs a Confirmation Density Algorithm (CDA) reactor trip safety function, which will be credited as part of the GGNS UFSAR, Chapter 7.6.1.5.6, "Average Power Range Monitor (APRM) Subsystem," for mitigation of a plant instability event. The OPRM also performs three other algorithms: period based detection algorithm (PBDA), amplitude-based algorithm (ABA) and growth-based algorithm (GRA), which are not credited safety functions, but are included as defense-in-depth features. The CDA function is used to demonstrate protection of the MCPR safety limit for anticipated reactor instabilities. A failure of the Nuclear Measurement Analysis and Control (NUMAC) OPRM or APRM could disable the automatic safety trip function performed by the DSS-CD algorithms. The GGNS NUMAC system includes a means of providing automatic backup stability protection (ABSP) in the event that the primary means of stability protection (i.e., DSS-CD) becomes inoperable. However; the NRC staff notes that use of common software for both primary (DSS-CD) and backup (ABSP) stability protection can lead to a condition where both of these automatic functions would become disabled due to a postulated software defect that could be triggered to result in a common-cause failure (CCF) of the OPRM reactor trip safety function.

If the OPRM system is inoperable, and the ABSP function performed by the APRM cannot be implemented or is also inoperable, manual backup stability protection (BSP) becomes the licensed stability solution. The **[[**

]]. When plant conditions exceed this BSP boundary and the plant ends up inside the Manual BSP Scram Region I, administrative actions require initiation of a manual reactor scram. This is described in Section 7 of the SAR and in the TS changes documented in the approved DSS-CD LTR, NEDC-33075P-A, Revision 6 (Reference 32).

Because of the potential for loss of both primary and backup automatic protection functions, the licensee performed a diversity and defense-in-depth (D3) analysis, which considered the effects of a postulated software CCF of the NUMAC power range neutron monitoring (PRNM) (APRM/OPRM) system in conjunction with the plant instability events described in the GGNS UFSAR. The results of this analysis were provided in the letter dated December 30, 2013 (Reference 2). **[[**

]]. This analysis identified manual operator actions as a diverse means of maintaining plant safety, if the automatic trip functions performed by the DSS-CD algorithms and the ABSP become unavailable due to a postulated common-mode failure of the NUMAC PRNM system.

The D-3 analysis identified that the postulated CCF in the PRNM system results in **[[**

]]. The GGNS procedures require immediate action to reduce reactor power in order to mitigate possible high growth-rate power oscillations following an unanticipated core flow reduction event. [[

]]. The D-3 analysis identified multiple, diverse [[
]] that are independent of the effects of the postulated PRNM system CCF.

When a 2RPT condition is identified, GGNS operators are procedurally required to insert a manual scram if the BSP boundary is exceeded. This immediate action is uncomplicated and was demonstrated on the GGNS simulator at a recent audit on October 15, 2014 (Reference 64). The whole "surprise" transient simulation took less than 5 seconds. The NRC staff confirmed that the systems used for the initiation of the manual scram and for confirmation that the scram was successful do not rely on digital or software-based technologies. The staff determined these systems would, therefore, not be affected by a postulated software CCF that renders the automatic protection functions inoperable.

[[

]] that are independent of the effects of the postulated PRNM system CCF.

[[

]]. However, GGNS operators are procedurally required to first identify if the plant has entered into a region of thermal hydraulic instability and then to insert a manual reactor scram if the BSP limits have been exceeded. The NRC staff confirmed the systems used for controlling core flow, reactor power, and manual scram do not rely on digital or software-based technologies. The staff determined these systems would, therefore, not be affected by a postulated software CCF of the PRNMS that renders the automatic protection functions inoperable.

In its previous evaluation [[ ]], the NRC staff concluded that the [[ ]] is an acceptable solution, because it provides sufficient protection against plant SLMCPR violations [[
]]. This evaluation further concludes the [[ ]] are sufficiently diverse from the digital PRNMS NUMAC systems. These systems provide an acceptable means of diverse protection for the DSS-CD safety function.

In the response to RAI-5 (Reference 7), GEH provided the preliminary GGNS BSP regions.

Proposed TSs have been provided to implement the change from Option III to DSS-CD. The specifications follow the standard industry practice and require restoration of the primary DSS-CD instrumentation within 120 days if both CDA and ABSP functions are inoperable. Also, within 90 days, a special report should be provided with a plan for restoration of the primary stability licensing option. ABSP provides acceptable stability protection while the primary DSS-CD option is declared inoperable. The NRC staff reviewed the proposed changes to the TSs to implement DSS-CD and found them acceptable.

Conclusion

The NRC staff also determined that the manual control measures **[[**
]] are sufficiently diverse from the digital PRNMS NUMAC systems, and therefore, provide an acceptable means of diverse protection for the DSS-CD safety function. The staff determined that the proposed license amendment allowing the plant operation in the MELLLA+ domain with DSS-CD provides reasonable assurance of adequate protection of public health, safety and security and on this basis, the staff finds the proposed license amendment acceptable.

The NRC staff reviewed the licensee's analyses related to the effects of the proposed operating domain extension on the thermal and hydraulic design of the core and the RCS. The staff concludes that the licensee has adequately accounted for the effects of the proposed operating domain extension on the thermal and hydraulic design and demonstrated that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to proven designs, (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The NRC staff further concludes that the licensee has adequately accounted for the effects of the proposed operating domain extension on the hydraulic loads on the core and RCS components. The NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of GDC 10 and 12 following implementation of the proposed operating domain extension. Therefore, the staff finds the proposed operating domain extension acceptable with respect to thermal and hydraulic design.

3.5. Emergency Systems

RS-001 (Reference 34) provides guidance for the review of emergency systems for operating domain extensions. Per the RS-001 the following systems must be reviewed:

1) Control rod drive system
2) Overpressure protection for the RCPB during power operation
3) Reactor core isolation cooling
4) Reactor heat removal system
5) SLCS

3.5.1. Control Rod Drive System

Regulatory Evaluation

The NRC staff's review covered the functional performance of the CRD system to confirm that the system can affect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents. The review also covered the CRDS cooling system to ensure that it will continue to meet its design requirements. The NRC's acceptance criteria are based on (1) GDC 4 insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC 23 insofar as it requires that the protection system be designed to fail into a safe state; (3) GDC 25 insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (4) GDC 26 insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (5) GDC 27 insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods to cool the core is maintained; (6) GDC 28 insofar as it requires the reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core; (7) GDC 29 insofar as it requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs; and (8) 10 CFR 50.62(c)(3) insofar as it requires that all BWRs have an alternate rod injection (ARI) system diverse from the reactor trip system, and that the ARI system has redundant scram air header exhaust valves. Specific review criteria are contained in SRP Section 4.6 (Reference 35(h)).

Technical Evaluation and Conclusion

The control rod design has not been modified relative to the baseline. The NRC staff concludes that the regulatory requirements in GDC 4, 23, 25, 26, 27, 28, 29, and 10 CFR 50.62(c)(3) continue to be satisfied by the design at GGNS.

3.5.2. Overpressure Protection for the RCPB During Power Operation

Regulatory Evaluation

Overpressure protection for the RCPB during power operation is provided by relief and safety valves and the RPS. The NRC staff's review covered relief and safety valves on the main steamlines and piping from these valves to the SP. The NRC's acceptance criteria are based on (1) GDC 15 insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs; and (2) GDC 31 insofar as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized under operating, maintenance, testing, and postulated accident conditions. Specific review criteria are contained in SRP Section 5.2.2.

Technical Evaluation and Conclusion

The licensee has evaluated the impact of the proposed operating domain extension on overpressure protection. The evaluation is documented in the GGNS SAR, Section 3.1 (Reference 28). The reactor pressure remains unchanged; therefore, the steam flow during normal operation or through a relief valve or break remains unchanged.

For GGNS, the limiting overpressure event is the MSIV closure followed by High-Flux Scram. Analyses in Section 3.1 of the GGNS SAR indicate that the peak vessel pressure remains unchanged, and it is below the ASME, Section III limit of 1375 psig limit. There is no change in overpressure relief capacity. The NRC staff, therefore, concludes that the requirements of GDC 15 and 31 continue to be met.

3.5.3. Reactor Core Isolation Cooling

Regulatory Evaluation

The RCIC system serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main FW system is isolated from the reactor vessel. In addition, the RCIC system may provide decay heat removal necessary for coping with an SBO. The water supply for the RCIC system comes from the condensate storage tank, with a secondary supply from the SP. The NRC staff's review covered the effect of the proposed MELLLA + on the functional capability of the system. The NRC's acceptance criteria are based on (1) GDC 4 insofar as it requires that SSCs important to safety be appropriately protected against dynamic effects; (2) GDC 5 insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions; (3) GDC 29 insofar as it requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs; (4) GDC 33 insofar as it requires that a system to supply reactor coolant makeup for protection against small breaks in the RCPB be provided so the specified acceptable fuel design limits are not exceeded; (5) GDC 34 insofar as it requires that an RHR system be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that SAFDLs and the design conditions of the RCPB are not exceeded; (6) GDC 54 insofar as it requires that piping systems penetrating primary reactor containment be designed with a capability to test periodically the operability of the isolation valves and to determine if valve leakage is within acceptable limits; and

(7) 10 CFR 50.63 insofar as it requires that the plant withstand and recover from an SBO of a specified duration. Specific review criteria are contained in SRP Section 5.4.6.

Technical Evaluation and Conclusion

The RCIC design has not been modified relative to the baseline, and the expanded operating domain does not have an impact on the gross thermal power. Thus, the NRC staff concludes that the requirements of GDC 4, 5, 29, 33, 34, 54, and 10 CFR 50.63 continue to be satisfied.

3.5.4. Residual Heat Removal System

Regulatory Evaluation

The reactor heat removal system is used to cool down the RCS following shutdown. The RHR system is typically a low-pressure system, which takes over the shutdown cooling function when the RCS temperature is reduced. The NRC staff's review covered the effect of the proposed MELLLA+ on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. The NRC's acceptance criteria are based on (1) GDC 4 insofar as it requires that SSCs important to safety be appropriately protected against dynamic effects; (2) GDC 5 insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions; and (3) GDC 34, which specifies requirements for an RHR system. Specific review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation and Conclusion

The RHR system design has not been modified relative to the baseline, and the expanded operating domain does not have an impact on the decay heat. Thus, the NRC staff concludes that the requirements of GDC 4, 5, 19, and 34 continue to be satisfied.

3.5.5. Standby Liquid Control System

Regulatory Evaluation

The SLCS provides backup capability for reactivity control independent of the control rod system. The SLCS functions by injecting a boron solution into the reactor to affect shutdown. The NRC staff's review covered the effect of the proposed MELLLA+ on the functional capability of the system to deliver the required amount of boron solution into the reactor. The NRC's acceptance criteria are based on (1) GDC 26 insofar as it requires that two independent reactivity control systems of different design principles be provided, and that one of the systems be capable of holding the reactor core subcritical under cold conditions; (2) GDC 27 insofar as it requires that the reactivity control systems be designed to a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions; and (3) 10 CFR 50.62(c)(4) insofar as it requires that the SLCS be capable of reliably injecting a borated water solution into the reactor pressure vessel at a boron concentration, boron enrichment, and flow rate that provides a set level of reactivity control. Specific review criteria are contained in SRP Section 9.3.5 and other guidance provided in Matrix 8 of RS-001 (Reference 34).

Technical Evaluation

The hot shutdown boron weight (HSBW) is calculated on generic basis for each fuel line (e.g., GNF2 in the case of GGNS). The HSBW is confirmed effective on plant- and cycle-specific basis with ODYN and TRACG ATWS calculations. The GGNS SAR, Section 9.3.1, documents these calculations. Both the licensing bases and the best estimate ATWS calculations show that the generic HSBW is effective to shut down the GGNS core under MELLLA+ initial conditions.

Because the peak pressure during ATWS with MELLLA+ has increased, the licensee has increased the SLCS pump discharge pressure in the TS from 1340 psig to 1370 psig. Due to this change, the licensee has increased the operability requirement for the SLCS pump piping and associated relief valve. The licensee indicated in an e-mail dated October 20, 2014 (Reference 11), that the SLCS piping is rated for 1700 psi, and the pump discharge relief valve setpoint is 1700 psi. Therefore, there is sufficient margin between the operating pressure of less than 1500 psig and the relief valve setpoint (1700 psig). As part of the MELLLA+ amendment, the licensee has proposed a TS change for the new relief valve setpoint of 1370 psig. Based on the above, the NRC staff concludes that the requested change is acceptable.

The SLCS design has not been modified relative to the baseline, and the SLCS boron inventory shutdown margin has been evaluated for the initial core in the GGNS SAR (Reference 28). TSs have been modified to ensure that the slightly larger ATWS peak pressure does not impede proper operation of the SLCS. Therefore, the NRC staff finds that sufficient information has been provided to review the SLCS, and the requirements of GDC 26 and 27 and 10 CFR 50.62(c)(4) continue to be satisfied.

Conclusion

The NRC staff reviewed the licensee's analyses related to the effects of the proposed MELLLA+ operating domain extension on the functional design of the SLCS. The staff concludes that the design has not been modified relative to the baseline. The regulatory requirements in GDC 4, 23, 25, 26, 27, 28, 29, and 10 CFR 50.62(c)(3) continue to be satisfied by the design.

The NRC staff reviewed the licensee's analyses related to the effects of the proposed MELLLA+ operating domain extension on the SLCS and concludes that the design has not been modified relative to the baseline, the slight ATWS peak reactor pressure increase requires increasing SLCS pump discharge pressure value in the TSs, and the SLCS boron inventory shutdown margin has been evaluated for the initial core in the GGNS SAR. Therefore, the licensee has adequately accounted for the effects of the proposed operating domain extension on the system and demonstrated that the system will continue to provide the function of reactivity control, independent of the control rod system, following implementation of the proposed MELLLA+ operating domain extension. Based on the above, the staff concludes that the SLCS will continue to meet the requirements of GDC 26, 27, and 10 CFR 50.62(c)(4) following implementation of the proposed MELLLA+ operating domain extension. Accordingly, the staff finds the proposed operating domain extension acceptable with respect to the SLCS.

3.6. NRC Evaluation of SAR Section 9.1, "Anticipated Operation Occurrences"

SAR Section 9.1, "Anticipated Operational Occurrences"

The licensee has performed a review of AOO transients and reported the results in Chapter 9.1 of the SAR.

Table 3 contains a summary of the AOO analysis evaluation. The AOOs analyzed in the SAR for the MELLLA+ domain extension include:

- Generator Load Rejection With No Bypass (LRNBP)
- Turbine Trip Without Bypass (TTNBP)
- Feedwater Controller Failure (Maximum Demand) (FWCF)
- Pressure Regulator Failure Downscales
- Loss of Feedwater Heater (LFWH)
- Control Rod Withdrawal Error (RWE)

The fuel loading error and the pressure regulator failure downscales are categorized as infrequent incidents for GGNS; therefore, they are not analyzed as an AOO.

As shown in Table 3, the remaining AOOs were evaluated at the CLTP and two core flows: the ICF limit of 105 percent and the MELLLA+ reduced core flow limit of 80 percent.

Table 3. Comparison of AOO Analyses Results at 80 Percent and 105 Percent Core Flow

Event	Parameter	Units	CLTP ICF (105%) Rated Core Flow	CLTP 80% Rated Core Flow
LRNBP				
	Peak Neutron Flux	% Initial	162.3	123.9
	Peak Heat Flux	% Initial	102.8	100.0
	Peak Vessel Pressure	psig	1,253.7	1,250.4
	Adjusted CPR Option B	NA	0.236	0.252
TTNBP				
	Peak Neutron Flux	% Initial	147.1	112.4
	Peak Heat Flux	% Initial	100.9	100.1
	Peak Vessel Pressure	psig	1,251.4	1,247.8
	Adjusted CPR Option B	NA	0.223	0.237

FWCF				
	Peak Neutron Flux	% Initial	119.3	105.5
	Peak Heat Flux	% Initial	104.0	103.4
	Peak Vessel Pressure	psig	1,242.8	1,238.1
	Adjusted CPR Option B	NA	0.202	0.192
LFWH				
	Adjusted CPR	NA	0.16	0.13

The operating limits to CPR and LHGR are adjusted upwards when operating at off-nominal conditions by power- and flow-dependent factors. The licensee has calculated the slow recirculation flow increase under MELLLA+ conditions to evaluate the power- and flow-dependent limits for a representative MELLLA+ equilibrium core. The results of these analyses are documented in Section 9.1.2 of the SAR. These results indicate that the existing limits in GGNS are adequate for MELLLA+ operation.

Non-limiting events in GGNS are treated via the generic disposition of these events in the MELLLA+ SER (Reference 30).

GNF2 Performance at Low Flows

The GGNS SAR calculations are based on a full equilibrium core of GNF2 fuel. Even though GNF2 **[[**

]]. Stability calculations are a special case where the parameter of merit is CPR performance at low-flow conditions. On a typical 2RPT, the CPR increases when the flow is reduced. Later in the transient, CPR is degraded if oscillations are established. The CPR increase due to the initial lower-to-flow reduction tends to dominate the final results on the analysis. A similar effect can be observed on the AOO analyses for MELLLA+ documented in Table 3. The low flow conditions (80 percent flow) tend to be limiting in these analyses, while the opposite is more common with fuels other than GNF2.

GEH simplified stability (GS3) is a newly approved stability long-term solution methodology. The GS3 methodology is intended to replace the Option I-D, II and III setpoint methodology (based on the old delta over initial MCPR versus oscillation magnitude) with **[[**

]]. During an NRC staff audit for the GS3 methodology (Reference 60), the staff reviewed a number of calculations and observed that the **[[**

]] . For all the cases the staff reviewed, a similar trend was observed. The figure below shows a calculation for the **[[**

]].

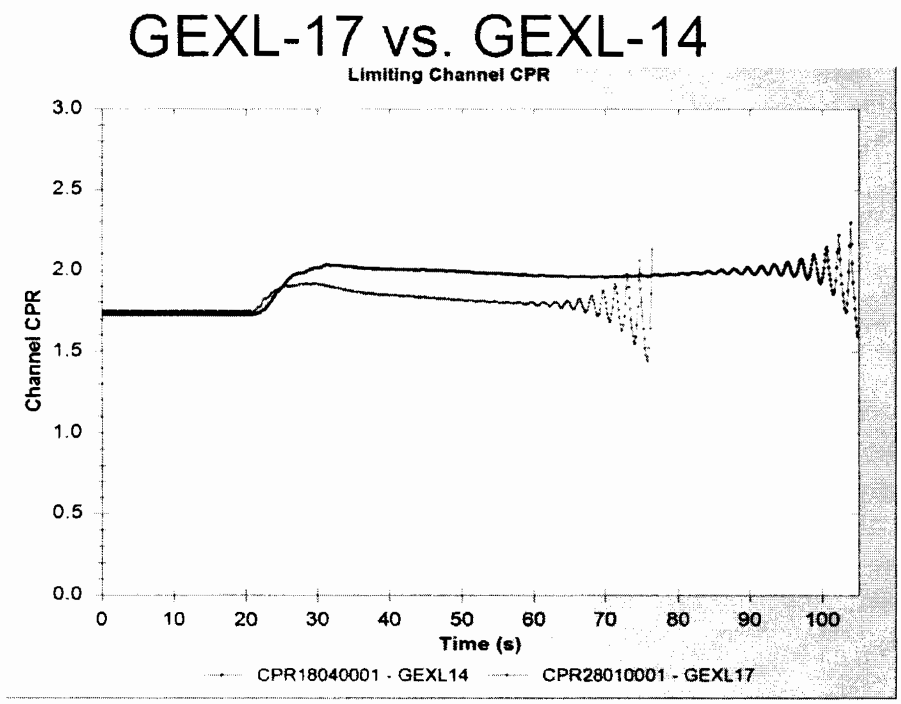
The NRC staff concludes that the **[[** **]]**, appears to be the case for the MELLLA+ results documented in Table 3.

Figure 4. Comparison of [[**]]**

[[

]]

Figure 5. Comparison of CPR Performance Following a 2RPT Using Two Different GEXL Correlations (GEXL14 – GE14, GEXL17 – GNF2)



3.7. NRC Evaluation of SAR Section 9.3, "Special Events"

The evaluation of MELLLA+ impact on special events is documented in Section 9.3 of the GGNS SAR. The SBO was evaluated generically in the MELLLA+ SER and the conclusions have been confirmed by the licensee in the GGNS SAR. The remaining special events are ATWS events, including ATWSI. The NRC satff's review of ATWS and ATWSI are presented below.

Regulatory Evaluation

ATWS is defined as an AOO followed by the failure of the reactor trip protection system specified in GDC 20. The regulation at 10 CFR 50.62 requires the following:

- Each BWR must have an ARI system that is designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

- Each BWR must have an SLCS with the capability of injecting into the reactor vessel a borated water solution with reactivity control at least equivalent to the control obtained by injecting 86 gallons per minute (gpm) of a 13 wt percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor vessel.

- Each BWR must have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS [for current-designed plants].

The NRC staff's review was conducted to ensure that (1) the above requirements are met, (2) sufficient margin is available in the setpoint for the SLCS pump discharge relief valve such that SLCS operability is not affected by the proposed MELLLA+, and (3) operator actions specified in the plant's EOPs are consistent with the generic emergency procedure guidelines/severe accident guidelines, insofar as they apply to the plant design. In addition, the staff reviewed the licensee's ATWS analysis to ensure that (1) the peak vessel bottom pressure is less than the ASME Service Level C limit of 1500 psig; (2) the PCT is within the 10 CFR 50.46 limit of 2200 °F; (3) the peak SP temperature is less than the design limit; and (4) the peak containment pressure is less than the containment design pressure. The staff also evaluated the potential for thermal-hydraulic instability in conjunction with ATWS events using the methods and criteria approved by the NRC staff. For this analysis, the staff reviewed the limiting event determination, the sequence of events, the analytical model and its applicability, the values of parameters used in the analytical model, and the results of the analyses.

Technical Evaluation

The licensee reviewed the family of ATWS events in the SAR (Reference 28). Based on this evaluation, the licensee concludes that the ATWS logic and setpoints remain unchanged for the proposed MELLLA+ operating domain extension; therefore, the limiting ATWS events are specified in the MELLLA+ SER:

1. Main steam isolation valve closure
2. Pressure regulator open failure
3. Loss of offside power

Two analysis methods are used: the licensing methodology, which uses ODYN, and a best estimate methodology, which uses TRACG04 with input data from TGBLA06/PANAC11. As required by the MELLLA+ SER limitation 12.18, the GGNS SAR lists the key operator actions credited, which include:

1. FW flow reduction starts within 30 to 45 seconds because of automated actions when the MSIV valves are closed, isolating motive steam for the feedwater pumps. When main steam is available for the FW pumps, 90 seconds are assumed for FW flow reduction.

2. Manual SLCS initiation within 300 seconds.

3. Initiation of RHR SPC within 660 seconds of event initiation.

For the licensing basis calculation (the ODYN calculation), the water level is controlled to 5 feet above the top of active fuel (TAF), and the SP is allowed to heatup, even after the heat capacity temperature limit (HCTL) is reached. For the best estimate calculation (TRACG), manual emergency depressurization is assumed within 40 seconds after the HCTL is reached.

With these assumptions, the peak vessel pressure is calculated by TRACG to reach 1300 psig, which is well below the 1500 psig ASME Service Level C limit. The calculations also show that MELLLA+ operation has a negligible effect on PCT and clad oxidation, because the peak channel power and limits remain unchanged.

The ODYN calculation indicates that, without depressurization, the SP temperature would reach a temperature of 197.5 °F, which is below the SP temperature design limit of 210 °F. The licensee notes that the licensing basis ODYN calculations for EPU conditions resulted in SP temperatures that violate the HCTL and would have required depressurization. Therefore, MELLLA+ does not present a new condition.

The best estimate TRACG calculations demonstrate that HCTL may or may not be reached, and emergency depressurization may not be required, but the outcome depends on the initial conditions. The HCTL is a function of the reactor operating pressure and the SP water level. For this reason, the licensee performed the best estimate analysis for bounding assumptions of HCTL from 139 °F to 151.4 °F. For the low HCTL value of 139 °F, depressurization is required. The temperature of 151.4 °F is the HCTL value at a pressure consistent with SRV lifting and is, thus, closer to the best estimate value for ATWS transients.

Section 9.3.1 of the GGNS SAR (Reference 28) presents the results of ATWS analyses. For all cases analyzed, the ATWS acceptance criteria are satisfied.

ATWS-I

The licensee has evaluated stability during ATWS events, and the results are documented in Section 9.3.1 of the SAR. The results of the ATWS instability analysis show that the mitigation actions in the GGNS EOP procedures (flow runback to uncover the FW spargers) will be effective in the MELLLA+ operating domain. The TRACG04 calculations indicate that all applicable fuel limits are satisfied for this event.

In the response to RAI-18, by letter dated August 26, 2014 (Reference 7), the licensee provided detailed plots of the channel powers, **[[**

]]. The relatively small oscillation amplitude is due to the prompt operator actions to reduce FW injection and vessel water level in <90 seconds. **[[**

]].

In the response to RAI-19 (by letter dated August 26, 2014 (see Appendix-A), the licensee provided more detailed plots of the PCT during the ATWS-I event. The most relevant data are reproduced here as Figure 6. **[[**

]].

Figure 6. ATWS-I. PCT Calculated [[

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[[

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The NRC staff has devoted a large effort to perform confirmatory ATWS-I calculations with the TRACE code. These confirmatory calculations showed results that did not necessarily agree with the TRACG04 results. The divergence between the codes occurred mainly because:

1. Different Tmin correlations are used by the two codes.
2. The quench models of both codes give significantly different results.

The following issues have been resolved:

1. The NRC staff reviewed experimental data provided by GEH to justify the use of the TRACG04 T_{min} correlation (Shumway with Zr credit) and found it acceptable for use in best estimate beyond design basis ATWSI calculations (see RAI-R1 response in the Appendix-A of this SER).

2. Requiring operator training to ensure that water level reduction is initiated within 90 seconds of the ATWS initiation. With this prompt water level reduction, the

oscillations do not grow as large as they could potentially, and T_{min} is never reached in GGNS, so the quench model is not required to demonstrate core coolability.

3. As the result of an RAI, a mistake in the original TRAC model formalism described in NUREG/CR-2178 was identified by GEH. This mistake was corrected in TRACG04, which makes the model consistent with the intended formulation and results in predicted axial conduction quench heat transfer consistent with the phenomena.

Because of the divergence observed between the two codes, the NRC staff requested in RAI-20 a comparison between the two codes using a common input deck. The TRACG04 and TRACE decks were made as similar as possible, and the results show relatively good agreement, considering the difficulty of the transient modeled and the fact that models and correlations between the codes are not exactly the same.

The detailed evaluation of RAI-20 is contained in Appendix A. The main observations from this evaluation are as follows:

1. When very unstable analysis conditions are assumed in the calculation (e.g., water level is maintained high longer), both the TRACE and TRACG codes predict that T_{min} is reached during prototypical ATWS-I events and exceeded with a prolonged heatup. The fuel heatup predicted in both cases indicates local fuel damage so that core coolability is potentially compromised.

2. For less stable conditions (e.g., with fast water level reduction and more stable initial flow and power conditions), both codes predict that the fuel clad temperature remains below T_{min} and core coolability is maintained.

The NRC staff conclusion from this evaluation is that both codes predict similar fuel response when using equivalent input parameters and the corrected quench model in TRACG04. Therefore, the TRACE calculations confirm the TRACG04 results when the quench model is not required ($T < T_{min}$). Thus, core coolability during ATWS-I has been demonstrated for GGNS. As a result of the TRACE confirmatory calculations, a coding mistake was corrected in TRACG04 that affected the quenching phenomena significantly. Based on the above data, the NRC staff concludes that the ATWS mitigation features and manual FW flow runback is adequate to mitigate the ATWS instability oscillations. The calculations indicate that ATWS acceptance criteria are satisfied, even in the presence of unstable power oscillations.

Conclusion

The NRC staff reviewed the information submitted by the licensee related to ATWS and concludes that the licensee has adequately accounted for the effects of the proposed MELLLA+ operating domain extension on ATWS. The NRC staff concludes that the licensee has demonstrated that ARI, SLCS, and manual FW runback systems have been installed, and that they will continue to meet the requirements of 10 CFR 50.62 and the analysis acceptance criteria following implementation of the proposed operating domain extension. Therefore, the staff finds the proposed operating domain extension acceptable with respect to ATWS.

3.8 Limitations from Applicable SERs

The GGNS SAR (Reference 28) appendices summarize the disposition of the limitations in the applicable SERs, including the following:

1. The Methods SER, NEDC-33173P-A (Reference 31)
2. The MELLLA+ SER, NEDC-33006P-A, Revision 3 (Reference 30)
3. The DSS-CD SER, NEDC-33075P-A (Reference 32)

Note that in prior MELLLA+ applications, a fourth appendix was included to account for one limitation of the TRACG application for DSS-CD (Reference 33). The new Revision 11 of DSS-CD SER incorporates the TRACG application, and the old limitation no longer applies. Therefore, a fourth appendix was not necessary.

Methods LTR NEDC-33173P-A Limitations

Appendix A of the GGNS SAR summarizes the disposition of limitations in the Methods SER, NEDC-33173P-A (Reference 31). The licensee states that the following Methods SER limitations do not apply to GGNS:

- 9.2, 3D Monicore – because the limitation is applicable only to TGBLA04/PANAC10 applications. The SAR is based on TGBLA06 and PANAC11.

- 9.4, SLMCPR 1 – superseded by Revision 4. For operation at MELLLA+, including operation at the EPU power levels at the achievable core flow state-point, a 0.01 value shall be added to the cycle-specific SLMCPR value for power-to-flow ratios up to 42 MWt/Mlbm/hr, a 0.02 value shall be added to the cycle-specific SLMCPR value for power-to-flow ratios above 42 MWt/Mlbm/hr.

- 9.13, Application of 10 Weight Percent Gd – because GGNS MELLLA+ uses GNF2 fuel, and as such does not seek to apply 10 wt percent Gd to this licensing application.

- 9.14, Part 21 Evaluation of GESTR-M Fuel Temperature Calculation – Part 21 report is related to GESTR-M T-M evaluation. GGNS evaluation is based on GNF2 fuel, which has a PRIME T-M model.

- 9.16, Void Reactivity 2, and 9.20, Void-Quality Correlation 2 – because the SAR uses void reactivity coefficients bias and uncertainties that are applicable to the GNF2 lattice designs loaded in the core as approved in Supplement 3 to NEDE-32906P, "Migration to TRACG04/PANAC11 from TACG02/PANAC10."

- 9.18, Stability Setpoints Adjustment – Not applicable to DSS-CD because the significant conservatisms in the current licensing methodology and associated MCPR margins are more than sufficient to compensate for the overall uncertainty in the OPRM instrumentation.

- 9.21, Mixed Core Method 1 – because GGNS MELLLA+ is not based on a mixed core.

- 9.22, Mixed Core Method 2 – because GNF2 is an approved fuel product line in the methods SER.

- 9.23, MELLLA+ Eigenvalue Tracking – This limitation in the MELLLA+ Methods LTR requires GEH to submit eigenvalue and power distribution tracking data following the first plant-specific implementation of MELLLA+. This limitation is not applicable to the licensee and the data will be used by GEH to validate the Methods LTR.

The disposition of the limitations applicable to GGNS is summarized on a table in Appendix A of the GGNS UFSAR and discussed in more detail in the body of the report. These limitations and their resolution are as follows:

- 9.1, TGBLA/PANAC Version – TGBLA06/PANAC11 methods are used.

- 9.3, Power-to-Flow Ratio – The GGNS MELLLA+ power density is 57.4 MWt/Mlbm/hr, which exceeds the 50 MWt/Mlbm/hr limit. The NRC staff resolution of this limitation involves two steps: (1) additional uncertainty is applied to the SLMCPR calculation by using the SLO flow uncertainties, and (2) plant-specific power distribution uncertainties have been evaluated based on GGNS TIP measurements.

- 9.5 SLMCPR 2 – The original condition has been superseded by the NRC staff SE for NEDC-33173P-A, Revision 4, dated November 2012 (Ref. 31). The conclusion of this SE stated that the original SLMCPR adders in the Methods SE are no longer applicable. Using the Revision 4 methodology, a 0.02 value shall be added to the cycle-specific SLMCPR value for points in the operating map with power/flow ratios greater than 42 MWt/Mlbm/hr. For points with power to flow ratio less than 42 MWt/Mlbm/hr, a 0.01 SLMCPR adder is used. The Revision 4 methodology does not change the SLO uncertainty penalty for SLMCPR evaluations in the MELLLA+ domain. In MELLLA+, SLO uncertainties are used to determine the acceptable SLMCPR, this is equivalent to 0.03 to 0.04 and hence is acceptable.

- 9.6, R-Factor – The R-factors are consistent with the axial void profiles expected in GGNS.

- 9.7, ECCS-LOCA 1 – The GGNS ECCS LOCA analyses include an evaluation for top-peaked and mid-peaked axial power profiles.

- 9.8, ECCS-LOCA 2 – The GGNS ECCS LOCA calculations have been performed at the MELLLA+ corner (100 percent CLTP 80 percent flow) and demonstrated compliance with limits.

- 9.9, Transient LHGR 1 and 9.10, Transient LHGR 2 – The SRLR was submitted to the NRC staff for review when available for the first cycle application. MELLLA+ limits are satisfied with the proposed operating margins for the MELLLA+ core.

- 9.11, Transient LHGR 3 – The results in Section 9 of the GGNS SAR demonstrate a 10 percent margin to T-M limits.

- 9.12, LHGR and Exposure Qualification – The GGNS SAR is based on PRIME.

- 9.15, Void Reactivity 1 – Void reactivity coefficients bias and uncertainties used in the latest version of TRACG are applicable to the GNF2 lattice designs loaded in the core.

- 9.17, Steady–State 5 Percent Bypass Voiding – Bypass voiding is conservatively estimated at 4.7 percent at the top of the TIP instrument, which satisfies the limitation.

- 9.19, Void-Quality Correlation 1 – The 0.01 OLMCPR penalty has been applied.

- 9.20, The GGNS M+ SAR licensing basis uses TRACG for ATWS-I analysis. The void reactivity coefficients bias and uncertainties used in the latest version of TRACG are applicable to the GNF2 lattice designs loaded in the core.

- 9.24, Plant-Specific Application – The bundle power, operating LHGR, and MCPR have been provided for the equilibrium GNF2 MELLLA+ GGNS cycle. All limits are satisfied.

MELLLA+ LTR NEDC-33006P-A Limitations

Appendix B summarizes the disposition of limitations in the MELLLA+ SER, NEDC-33006P-A,. Revision 3 (Reference 30). The licensee states that the following methods of SER limitations do not apply to GGNS:

- 12.10.c, ECCS-LOCA Off-rated Multiplier – Because GGNS MELLLA+ takes credit for off-rated limits at minimum core flow state point; therefore, core monitoring is required per limitation 12.10.d.

- 12.20, Generic ATWS Instability – Because GGNS does not use the generic ATWS stability analysis and has performed a plant-specific ATWS instability evaluation.

The disposition of the limitations applicable to GGNS is summarized on a table in the GGNS SAR Appendix B and discussed in more detail in the body of the report. These limitations and their resolution are as follows:

- 12.1, GEXL-Plus – GEXL-Plus applicability has been confirmed in Section 1.1.3 of the SAR.

- 12.2, Related LTRs – The limitations from NEDC-33173P-A, and NEDC-33075P-A are specifically addressed in Appendices A and C of the SAR. Limitations of NEDC-33147 are not addressed since TRACG is now approved for DSS-CD stability solution calculations.

- 12.3, Concurrent Changes

- 12.3.a, As addressed in Section 1.1.2 of the SAR, concurrent changes have been taken into account in the evaluation.

- 12.3.b, As addressed in Section 1.1.1 of the SAR, all generic dispositions have been reviewed for applicability.

- 12.3.c, As addressed in Section 1.1.1 of the SAR, generic bounding sensitivities have been reviewed for applicability.

- 12.3.d, ATWS instability analyses supporting the MELLLA+ condition are based on GGNS fuel response.

- 12.3.e, GNF2 was approved for expanded operating domain in Supplement-3 of methods LTR NEDC-33173P-A (Reference 29), and new analyses were performed for a specific core configuration.

- 12.3.f, GGNS will have a full load of GNF2 fuel. The DSS-CD resolution has been updated with GNF2 analysis. Conditions have been met.

- 12.3.g, DSS-CD will be employed in GGNS to address possible instabilities. DSS-CD has been approved for MELLLA+ applications.

- 12.4, Reload Analysis Submittal – The GGNS application provided the plant-specific thermal limits and transient assessment in the SRLR (Reference 42). The COLR is expected to be submitted before the approval (to be confirmed).

- 12.5, Operating Flexibility

- 12.5.a, SLO operation is not allowed in MELLLA+. The GGNS TSs will be revised as part of this amendment.

- 12.5.b, FWHOOS is not allowed in GGNS under MELLLA+.

- 12.5.c, The licensee has committed to provide the power-flow map in the COLR (see 12.4 above).

- 12.6, SLMCPR State Points and CF Uncertainty – The licensee has evaluated the SLMCPR at off-nominal conditions, including the 55 percent flow statepoint, and has reported it in the SRLR.

- 12.7, Stability – The DSS-CD automated backup stability option will be implemented at GGNS.

- 12.8, Fluence – By letter dated August 18, 2015 (ADAMS Accession No. ML15229A218), the NRC issued Amendment No 204 for GGNS. In this amendment, the NRC staff completed its review of Entergy's request to adopt a single fluence methodology from 0 effective full power years through the end of extended operations under EPU conditions. Based on this review, the NRC staff has concluded the effect of the change in the fluence while operating in the MELLLA+ domain will have acceptable effects on plant components.

- 12.9, Reactor Coolant Pressure Boundary – A discussion of non-category-A materials is presented in the SAR, Section 3.5.1.4. An augmented inspection program has been implemented at the plant, and the NRC staff concludes that the augmented inspection program at GGNS is adequate to address IGSCC concerns related to "other than Category A" materials in the RCPB. This topic is further discussed in Section 3.2, SAR Section 3.5.1.4, "Other than Category 'A' RCPB Material," of this SE.

- 12.10, ECCS-LOCA Off-rated Multiplier

- 12.10.a, GGNS-specific Appendix K ECCS LOCA calculations were provided in the SAR. The PCT results are determined to be bound by the high-flow PCT values.

- 12.10.b and d, GGNS has opted for monitoring off-rated LOCAL limits and taking credit for them during the analysis. The licensee has committed to confirm these off-rated limits for every future reload.

- 12.11, ECCS-LOCA Axial Power Distribution Evaluation – Top-peaked and mid-peaked power shapes have been used for the LOCA analyses.

- 12.12, ECCS LOCA Reporting

- 12.12.a and 12.12.b, Both the nominal and the Appendix K LOCA results have been reported in the SAR. The current uncertainty method was used.

- 12.13, Small Break LOCA and 12.14, Break Spectrum – No small break LOCA PCT calculations were required for GGNS, because the rated small break PCT is significantly lower than the Appendix K values. A number of small break sizes were evaluated to determine the limiting event for the GGNS EPU application.

- 12.15, Bypass Voiding above the D-Level – Bypass voiding has been calculated to be 4.7 percent at the top of the TIP instrument, which is lower than the 5 percent limit.

- 12.16, RWE – A plant-specific RWE analysis was performed using PANACEA to confirm the validity of the RWL setpoints.

- 12.17, ATWS Loop – ATWS calculations were performed in the SAR, Section 9.3.1 using the licensing basis (ODYN) and a best estimate code (TRACG).

- 12.18, ATWS TRACG Analysis

- 12.18.a, 12.18.b, and 12.18.c, TRACG ATWS calculations were performed to demonstrate compliance with ATWS criteria because (1) the licensing bases ODYN calculation showed that HCTL limit would be reached, and (2) the licensee opted not to increase the boron 10 concentration.

- 12.18.d, The GGNS TS LCOs will implement the limitation of no FWHOOs and no SLO operation in the MELLLA+ region. An evaluation has been made on the GGNS SAR that the number of SRVs that must be available is unchanged.

- 12.18.e and 12.18.f, The key assumptions used for the ATWS analyses and the treatment of uncertainties are documented in GGNS SAR Section 9.3.1.

- 12.19, Plant-Specific ATWS Instability – The licensee has provided a best estimate ATWS stability calculation using TRACG04 to demonstrate compliance with limits.

- 12.21, Individual Plant Evaluation – A plant-specific probabilistic risk assessment was included in the GGNS SAR, Section 10.5. Based on these analyses, the licensee concludes that the risk increase lies within Region III (i.e., changes that represent very small risk changes).

- 12.22, IASCC – (Irradiation Assisted Stress-Corrosion Cracking) Fluence calculations indicate that the top guide, core plate, and shroud exceed the threshold. The inspection strategies in place are considered sufficient As discussed in limitation 12.8, the NRC staff completed its review of Entergy's request to adopt a single fluence methodology from 0 effective full power years through the end of extended operations under EPU conditions. As part of the fluence LAR, the NRC staff concluded that operation in the MELLLA+ domain will not have a large effect on the fluence values.

- 12.23, Limitations from the ATWS RAI Evaluations

- 12.23.1, See Limitation 12.18.d.

- 12.23.2, The ATWS calculations key parameters were provided.

- 12.23.3, The SRV tolerances were included in the ATWS analyses.

- 12.23.4, The EOP procedures were reviewed and sensitivity analyses performed for different water level control strategies. The EOPs require the operator to lower level to top of active fuel (TAF) (unless the transient terminates early) and control within a band between the minimum steam cooling water level and 2 feet below the spargers. A wide band is necessary because manual level control during an ATWS cannot be accomplished accurately. The sensitivity calculations indicate that the EOP strategy is adequate to satisfy the ATWS criteria.

- 12.23.5, The GGNS MELLLA+ power density at the full-power minimum flow statepoint is 49.0 MWt/Mlbm/hr, which does not exceed the 52.5 MWt/Mlbm/hr limit.

- 12.23.6, ATWS Instability analysis was performed for GNF2 fuel.

- 12.23.7, See 12.23.6, above.

- 12.23.8, The ATWS calculations accounted for all GGNS-specific features.

- 12.23.9, The plant-specific ATWS calculations accounted for the physical limitations of ECCS systems used (RCIC in the GGNS case).

- 12.23.10, The containment pressure calculated by the best estimate TRACG analysis is 6.0 psig given in the SAR, Table 9-8, which is under the containment limit of 15 psig for GGNS. All safety grade equipment will function under this containment overpressure condition.

- 12.23.11, The HCTL values used for ATWS calculations are the nominal vales. They are a function of vessel pressure and SP water level.

- 12.24, Limitations from Fuel-Dependent Analyses RAI Evaluations

- 12.24.1, The TRACG GGNS-specific calculations model the water rod flow explicitly.

- 12.24.2, The core exit void fraction is presented in Table 1-2 of the SAR for a MELLLA and MELLLA+. The highest void fraction under MELLLA+ corresponds to the low flow point (80.6 percent CLTP, 55 percent flow) and has a value of 77 percent, compared to 73 percent for the nominal MELLLA condition (100 percent CLTP, 92.8 percent flow).

NEDC-33075P-A Limitations

Appendix C summarizes the disposition of limitations in the DSS-CD LTR, NEDC-33075P-A, Revision 8 (Reference 32). The disposition of the limitations applicable to GGNS is summarized on a table in Appendix C of the GGNS SAR and discussed in more detail in the body of the report. These limitations and their resolution are:

- 5.1 – The DSS-CD will be implemented in the already approved GE Option III platform.

- 5.2 – The DSS-CD Confirmation Density Algorithm setpoint calculation followed the procedure outlined in the DSS-CD LTR NEDC-33075P-A (Reference 32).

- 5.3 – The values of the FIXED and ADJUSTABLE parameters are established by GEH and will be documented in a DSS-CD Settings Report.

- 5.4 – V&V of the DSS-CD trip function code was performed for transportability considerations.

It must be noted that the previous version of DSS-CD LTR, NEDC-33075P-A, Revision 6, relied on a separate LTR, NEDE-33147P-A (Reference 33), for the TRACG04 application. When Revision 7 of the DSS-CD LTR was issued, it incorporated the TRACG04 application, and the limitations of that LTR are no longer applicable.

The NRC staff concurs with the licensee evaluation of the above limitations in the listed SERs.

3.9 Use of TRACG

The NRC staff reviewed the TRACG code models and concludes that TRACG calculations of ATWS-I for GGNS with possible rewetting and quenching is sufficient to provide reasonable assurance of compliance with the applicable ATWS regulatory criteria, namely, demonstrating that core coolability is maintained during ATWS-I events. This staff review considered plant-specific information (e.g., EOPs), specific aspects of TRACG code use as it was applied in the context of the GGNS ATWS-I analysis provided by the licensee (e.g., updates to the quench model, revision to the T_{min} correlation in TRACG, etc.), and justification of the applicability of experimental data. The current review does not constitute a generic review and approval of the TRACG method.

With the proposed prompt operator actions (< 90 seconds to reduce FW flow), **[[**

]].

The GGNS is a BWR/6 with a Mark III pressure suppression type primary containment consisting of a reinforced concrete right circular cylinder with a hemispherical domed roof and a flat base slab. The containment includes a DW, a WW, and the region above the WW HCU floor. The DW is a cylindrical reinforced concrete structure, which surrounds the reactor vessel and its support structure. The lower portion of the DW wall is submerged in the SP. Three rows of circular vents, 45 per row, penetrate the DW wall below the normal level of the SP. The WW, which includes the SP, is considered to be a portion of the containment. The SP weir wall located inside the DW acts as the inner boundary of the SP. It is constructed of reinforced concrete and extends from the outer edge of the DW sump floor. The weir wall is lined with stainless steel. The WW-to-DW vacuum relief system consists of two vacuum breakers, which equalize the pressure between the containment and the DW to prevent a backflow of water from the SP into the vent system.

4.0 OPERATING LICENSE AND TECHNICAL SPECIFICATION CHANGES

To achieve the MELLLA+, the licensee proposed the following changes to the facility operating license and TSs for GGNS.

4.1. Proposed License Conditions

4.1.1. FWHOOS Limitation: New License Condition 2.C.(48)

The licensee has proposed to add license condition 2.C.(48). The new license condition prohibits operating with a FWHOOS while in the MELLLA+ domain, and would state:

> (48) Feedwater Heaters Out-of-Service (FWHOOS)
>
> Operation with FWHOOS in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) region is prohibited.

NRC Staff Evaluation

This is a required condition for operating in the MELLLA+ domain by the MELLLA+ SER (Limitation 12.5.b), and therefore, is acceptable. This is discussed in Section 1.3 of this SE.

4.1.2. Operator Training: New License Condition 2.C.(49)

The licensee has proposed to add license condition 2.C.(49). The new license condition, which requires testing and training of the operators for critical operator actions, would state:

(49) Time-Critical Operator Action Commitments made as required for the MELLLA+ LAR will be converted to a License Condition as follows:

Prior to Operation in the MELLLA+ Domain, Entergy will:

Train all active operating crews to perform the following three MELLLA+ time-critical operator actions:

1. Initiate Reactor Water Level Reduction (90 seconds following failure to scram concurrent with no reactor recirculation pumps in service and CTP> 5%).
2. Initiate Standby Liquid Control Injection (300 seconds if CTP> 5% or before Suppression Pool Temperature reaches 110 degrees F).
3. Initiate Residual Heat Removal Suppression Pool Cooling (660 seconds).

GGNS will validate that all active operating crews have met the time requirements for the three MELLLA+ time-critical operator actions during evaluated scenarios.

GGNS will report any MELLLA+ time-critical actions that are converted to "immediate actions" to the NRC Project Manager.

The following are one-time actions, which expire after the first report:

The results of the three MELLLA+ time-critical operator actions training will be reported to the NRC Project Manager within 60 days of completion of the training.

The reported results will include the full range of response times for each time-critical action and the average times for each crew.

Any MELLLA+ time-critical operator training failures during evaluated scenarios will be reported to the NRC within 60 days of any failures with a plan for resolution.

NRC Staff Evaluation

The licensee discussed the critical operator actions in Section 10.4 of the GGNS SAR. The review of the critical operator actions, this license condition and the basis for approval are discussed in Sections 1.3, 3.3, and 3.7 of this SE.

4.2. Technical Specification Changes for Implementation of DSS-CD

The licensee submitted changes to the GGNS TSs to support its MELLLA+ LAR. The proposed TS changes below are associated with implementation of the DSS-CD long-term stability solution and described in NEDC-33075P-A, "Licensing Topical Report, General Electric Boiling Water Reactor Detect and Suppress Solution-Confirmation Density" Revision 8, dated November 19, 2013 (Reference 32), Section 8.0, "Effect of Technical Specifications." The NRC staff's review of the proposed changes is discussed below.

4.2.1. TS 3.3.1.1 Reactor Protection System Instrumentation, Conditions J and K, and New Condition L

Current TS 3.3.1.1 Required Action J and K states:

J.1 Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.

AND

J.2 -------------------------------NOTE---
LCO 3.0.4 is not applicable.

Restore required channels to OPERABLE.

The current Completion Times for J.1 is 12 hours and for J.2 is 120 days.

K.1 Reduce THERMAL POWER TO < 21% RTP.

The current Completion time for K.1 is 4 hours.

TS 3.3.1.1 Required Actions J and K were replaced with:

J.1 Initiate action to implement the Manual BSP Regions defined in the COLR.

AND

J.2 Implement the Automated BSP Scram Region using the modified APRM Flow Biased Simulated Thermal Power – High trip function setpoints defined in the COLR.

AND

J.3 Initiate action to submit an OPRM report in accordance with Specification 5.6.7.

The new Completion Time for J.1 is Immediately, for J.2 is 12 hours and for J.3 is Immediately.

K.1 Initiate action to implement the Manual BSP Regions defined in the COLR.

AND

K.2 Reduce operation to below the BSP Boundary defined in the COLR.

AND

K.3 --------------------------NOTE---
LCO 3.0.4.a is not applicable.

Restore required channels to OPERABLE.

The new Completion Time for K.1 is immediately and for K.2 is 12 hours and for K.3 is 120 days.

New Condition L will state:

L. Required Action and associated Completion Time of Condition K not met.

New Required Action L.1 will state:

L.1 Reduce THERMAL POWER TO < 16.8% RTP.

The Completion Time for new Condition L will be 4 hours.

The licensee proposed to replace the Action Statements and Completion Times for current TS 3.3.1.1 Conditions J and K and to insert a new Condition L to support implementation of the BSP requirements in the event that DSS-CD is inoperable. Condition J is the same with 3 new Required Actions and Completion times instead of two. Condition K is the same with 3 new Required Actions and Completion Times instead of 1. The NOTE added to K.3 is consistent with the NOTE previously used in current J.2. New Condition L has been added which consistent with current Condition K. The new Required Action is:

Reduce THERMAL POWER TO < 16.8% RTP.

Current Required Action K states "Reduce THERMAL POWER TO < 21 % RTP." Section 3.5 of the DSS-CD LTR (Reference 32) requires DSS-CD to be operable above a power level set at

5 percent below the lower boundary of the armed region defined by the MCPR threshold power level, which for GGNS is 21.8 percent (see SAR Section 2.4.2). The proposed changes are consistent with the requirements specified in Section 8.0 of Reference 32 and are required for the implementation of DSS-CD. Therefore, the NRC staff concludes that these changes are acceptable.

4.2.2. TS 3.3.1.1, Reactor Protection System Instrumentation (RPS) Instrumentation, Surveillance Requirement (SR) 3.3.1.1.23

The licensee proposed to delete SR 3.3.1.1.23. The surveillance is no longer required and eliminates unnecessary actions. The proposed changes are consistent with the requirements specified in Section 8.0 of Reference 32. As this deletion is required for the implementation of DSS-CD, the NRC staff concludes that this change is acceptable.

4.2.3. TS 3.3.1.1, Reactor Protection System Instrumentation, Table 3.3.1.1-1

The licensee proposed to add new Note (g) to Table 3.3.1.1-1, Function 2.d to require resetting the allowable value setpoint for the Flow Biased Simulated Thermal Power - High when the OPRM is inoperable. New Note (g) will read as follows:

> (g) With the OPRM Upscale trip function (function 2.f) inoperable, reset the APRM-Flow Biased Simulated Thermal Power – High trip function (Function 2.d) setpoints to the values defined by the COLR to implement the Automated BSP Scram Region in accordance with Action J of this specification.

The proposed change is consistent with the requirements specified in Section 8.0 of Reference 32. As this new note is required for the implementation of DSS-CD, the NRC staff concludes that this change is acceptable.

4.2.4. TS 3.3.1.1, Reactor Protection System Instrumentation, Table 3.3.1.1-1, Function 2.f

Current TS 3.3.11, Table 3.3.1.1-1, Function 2.f, Applicable Modes or Other Specified Conditions is ≥ 21%.

Revised TS 3.3.1.1, Table 3.3.1.1-1, Function 2.f, Applicable Modes or Other Specified Conditions will be ≥ 16.8% RTP.

In Table 3.3.1.1-1, Function 2f, Surveillance Requirement 3.3.1.1.23 will be deleted.

Consistent with TS 3.3.1.1, the licensee proposed to change the value in the Applicable Modes or Other Specified Conditions column of Table 3.3.1.1-1 from ≥ 21 percent to ≥ 16.8 percent RTP. Section 3.5 of the DSS-CD LTR (Reference 32) requires DSS-CD to be operable above a power level set at 5 percent below the lower boundary of the armed region defined by the MCPR threshold power level, which for GGNS is 21.8 percent (see SAR Section 2.4.2).

Therefore, DSS-CD must be operable at 16.8 percent (21.8 percent – 5 percent). This change is consistent with the plant-specific analyses (Section 8 of Reference 32), and is required for the implementation of DSS-CD. Based on the above, the NRC staff concludes that this change is acceptable.

4.2.5. TS Section 3.3.1.1, RPS Instrumentation, Table 3.3.1.1-1, Note (b)

The current TS 3.3.1.1, Table 3.3.1.1-1, Note (b) states:

(b) Two-Loop Operation: 0.58W + 59.1% RTP and ≤ 113% RTP
Single-Loop Operation: 0.58W + 37.4% RTP

Revised TS 3.3.11, Table 3.3.1.1-1, Note (b) will state:

(b) Two-Loop Operation: 0.64W + 61.8% RTP and ≤ 113% RTP
Single-Loop Operation: 0.58W + 37.4% RTP

The licensee proposed to revise the allowable value for Function 2.d, "Flow Biased Simulated Thermal Power – High" (for two-loop operation) from 0.58W+ 59.1 RTP to 0.64W + 61.8 percent RTP. The basis for the allowable value setpoint is discussed in SAR Section 5.3.1, "APRM Flow-Biased Scram." The proposed change maintains the margin between the MELLLA+ operating domain and the current trip. The proposed change is consistent with the requirements specified in Section 8.0 of Reference 11 and required for the implementation of DSS-CD. Based on the above, the NRC staff concludes the change is acceptable.

4.2.6. TS Section 3.3.1.1, Reactor Protection System Instrumentation (RPS) Instrumentation, Table 3.3.1.1-1, Note (f)

Current TS 3.3.1.1, Table 3.3.1.1.-1, Note (f) states:

(f) The setpoint for the OPRM Upscale Period-Based Detection algorithm is specified in the COLR.

Revised TS 3.3.1.1, Table 3.3.1.1-1, Note (f) will state:

(f) The setpoint for the OPRM Upscale Confirmation Density Algorithm (CDA) is specified in the COLR.

The licensee proposed to revise Note (f) to reflect the setpoints for the OPRM Upscale from the Periodic Based Detection Algorithm to the Confirmation Density Algorithm. This change reflects the change from Option III stability solution to DSS-CD. In addition, this change is required to be consistent with the COLR designations. Based on the above, the NRC staff concludes that this change is acceptable.

4.2.7. TS 3.4.1, Recirculation Loops Operating

Current TS LCO 3.4.1 states, in part:

> One recirculation loop shall be in operation with the required limits modified for single loop operation as specified in the COLR.

Revised TS LCO 3.4.1 will state, in part:

> One recirculation loop shall be in operation provided the plant is not operating in the MELLLA+ domain defined in the COLR and provided the required limits are modified for single loop operation as specified in the COLR.

The licensee proposed to modify TS LCO 3.4.1, "Recirculation Loops Operating." The proposed change further defines requirements while in SLO, and restricts SLO in the MELLLA+ operating domain. Operation in the MELLLA+ domain is not analyzed for SLO.

As discussed in the GGNS SAR Section 3.6.3, SLO is not allowed in the MELLLA+ operating domain. The proposed modification to LCO 3.4.1 recognizes that one recirculation loop may be in operation provided the plant is not operating in the MELLLA+ operating domain as defined in the COLR. The NRC staff reviewed the proposed change and concludes the proposed change is consistent with the SAR and required for the implementation of DSS-CD, and therefore, concludes it is acceptable.

4.2.8. TS 5.6.5, Core Operating Limits Report (COLR)

Current TS 5.6.5.a.6 states:

> Deleted

Revised TS 5.6.5.a.6 will state:

> 6) The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Flow Biased Simulated Thermal Power – High trip function (Function 2.d setpoints used in the OPRM, Automated BSP Scram Region, and BSP Boundary for TS 3.3.1.1.

The proposed change to TS 5.6.5.a.6 is consistent with the requirements specified in Section 8.0 of Reference 32. The changes will ensure that applicable thermal limits continue to be met and reflect NRC-approved analytical methodologies. As the proposed change is required for the implementation of DSS-CD, the NRC staff concludes that this change is acceptable.

In a letter dated January 6, 2015 (Reference 16), the licensee proposed revision of TS 5.6.5, COLR, to add NEDC-33075P-A, "Detect and Suppress Solution-Confirmation Density Licensing Topical Report." The applicability of this LTR for GGNS for the current and future COLR reviews has been confirmed by this review. Entergy has requested in a separate request that the NRC staff add this LTR as an approved topical report for COLR analysis.

4.2.9. TS 5.6.7, Oscillation Power Range Monitor (OPRM) Report

The licensee proposed to add a new item to TS 5.6., "Reporting Requirements," as follows:

> 5.6.7 Oscillation Power Range Monitor (OPRM) Report
>
> When an OPRM report is required by Condition J of LCO 3.3.1.1, "RPS Instrumentation," it shall be submitted within the following 90 days. The report shall outline the preplanned means to provide backup stability protection, the cause of the inoperability, and the plans and schedule for restoring the required instrumentation channels to OPERABLE status.

This change is consistent with the requirements specified in Section 8 of Reference 11 and conforms to the content and structure of structure of the TSs as described in NUREG-1433. Based on the above, and as the change is required for the implementation of DSS-CD, the NRC staff concludes that this change is acceptable.

As noted above, the above changes are required by the DSS-CD LTR for implementation of the DSS-CD. Solution. The NRC staff has confirmed that the proposed TS changes are consistent with the TS changes proposed in the DSS-CD LTR and therefore, are acceptable.

4.2.10. Other TS Changes

4.2.10.1 TS SR [Surveillance Requirement] 3.1.7.7, SLCS Pump Test

Current SR 3.1.7.7 states:

> Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure of ≥ 1340 psig.

Revised SR 3.1.7.7 will state:

> Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure of ≥ 1370 psig.

The licensee proposed to revise the SLCS pump discharge pressure from 1340 psig to 1370 psig. Because the peak pressure during ATWS has increased, the licensee needed to increase the SLCS pump discharge pressure in the TSs from 1340 psig to 1370 psig. The licensee indicates that the SLCS piping is rated for 1700 psi, and the pump discharge relief valve setpoint is 1700 psi. Since there is a minimal decrease in the margin between the operating pressure and the relief valve setpoint, the NRC staff concludes that the requested change is acceptable.

4.2.10.2 TS 5.5.12, Appendix J, Testing Program

The licensee proposes to change the calculated peak containment internal pressure (P_a) from 14.8 psig to 12.1 psig. This change was reviewed by the NRC staff in Section 3.3, "Plant-Specific Dispositions," of this SE; and SAR, Section 4.1.1, "Short-Term Pressure and Temperature Response," and found to be acceptable. Based on the above, the NRC staff concludes that the proposed change is acceptable.

4.3. Regulatory Commitments

By letter dated September 25, 2013 (Reference 1), Entergy made the following regulatory commitments:

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Upon implementation of the TS amendment associated with approval of the MELLLA+ LAR, the DSS-CD algorithm will be enabled.	✓		90 days from receiving approval of the MELLLA+ LAR.
GGNS will include the power/flow map in the COLR after the MELLLA+ operating domain expansion is approved.		✓	90 days from receiving approval of the MELLLA+ LAR.
Any increase of moisture content above the design limit of 0.10 wt. % will be evaluated for effect on the FAC monitoring program.		✓	90 days from receiving approval of the MELLLA+ LAR.
Testing will be performed near the CLTP and the MELLLA+ minimum core flow state point of 80% as well as other state points that may be deemed valuable for the purpose of defining the MCO magnitude and trend.	✓		90 days from receiving approval of the MELLLA+ LAR.
Required changes are part of the MELLLA+ implementation plan and will be made consistent with the licensee's current plant training program requirements. These changes will be made consistent with similar changes made for other plant modifications and include any changes to TS, EOPs, and plant systems.	✓		90 days from receiving approval of the MELLLA+ LAR.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Consistent with the requirements for the plant-specific analysis as described in the M+LTR, the operator training program and plant simulator will be evaluated to determine the specific changes required. Simulator changes and fidelity validation will be performed in accordance with applicable American National Standards Institute (ANSI) standards currently being used at the training simulator.	✓		90 days from receiving approval of the MELLLA+ LAR.
Training required to operate GGNS following the MELLLA+ operating domain expansion will be conducted prior to operation in the MELLLA+ domain.	✓		90 days from receiving approval of the MELLLA+ LAR.
Training for the MELLLA+ startup testing program will be performed using "just in time" training of plant operation personnel where appropriate.	✓		90 days from receiving approval of the MELLLA+ LAR.
Enhanced training on ATWS event mitigation in the MELLLA+ domain will be conducted.	✓		90 days from receiving approval of the MELLLA+ LAR.
In accordance with M+LTR SER Limitation and Condition 12.23.4, the EOPs will be reviewed for any effect and revised as necessary prior to implementation of the MELLLA+ operating domain expansion. Any changes identified to the EOPs will be included in the operator training to be conducted prior to implementation of MELLLA+.	✓		90 days from receiving approval of the MELLLA+ LAR.
The AOPs will be reviewed for any effect and revised as necessary prior to implementation of the MELLLA+ operating domain expansion. Any changes identified to the AOPs will be included in the operator training to be conducted prior to implementation of MELLLA+.	✓		90 days from receiving approval of the MELLLA+ LAR.

The NRC staff concludes that reasonable controls for the implementation and subsequent evaluation of proposed changes pertaining to the above regulatory commitments are best provided by the licensee's administrative processes, including its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes).

4.4. Conclusion

The NRC staff reviewed the proposed amendment for GGNS to operate in the MELLLA+ domain, as documented in the GGNS SAR (Reference 28). The NRC staff reviewed the licensee's analyses related to the effect of the proposed extension on the operation of the GGNS. The staff concludes from this review that the broadening of the GGNS operating domain by lowering the flow at high powers without additional limitations would reduce the safety margin. However, the licensee has proposed solutions in the SAR that are technically acceptable to satisfy the regulatory criteria while operating in the MELLLA+ domain. The following proposed solutions will maintain the safety margin under the MELLLA+ domain the same as under the current operating domain:

1) FWHOOS operation will not be allowed in the MELLLA+ domain, because analyses have not been performed to demonstrate compliance with applicable criteria under these conditions.

2) SLO is not allowed in the MELLLA+ domain.

3) To provide additional protection against spurious, noise-induced scrams on the DSS-CD system, [[

]] the TLO and SLO MCPR Margin criteria documented in Tables 2-4 and 2-5 of the SAR are satisfied [[

]] the process described in Section 2.4 of the SAR and in Section 6 of the DSS-CD LTR (Reference 32).

The NRC staff concludes that for most of the systems operating in the MELLLA+ domain, there is no effect on the operation of GGNS. Specifically, the licensee has adequately accounted for the effects of the proposed MELLLA+ operating domain extension on the nuclear design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients. Further, the effects of postulated reactivity accidents will not cause significant damage to the pressure vessel or impair the capability to cool the core. Based on the above, the NRC staff concludes that the licensee, in its SAR, has adequately addressed the issues identified in the MELLLA+ LTR and the plant will continue to meet the applicable RGs, SRPs, and regulatory requirements of GDCs. Therefore, the NRC staff concludes that the proposed MELLLA+ operating domain extension is acceptable.

5.0 RECOMMENDED AREAS FOR INSPECTION

As described above, the NRC staff conducted an extensive review of the licensee's plans and analyses related to the proposed MELLLA+ implementation and concluded that they are acceptable. The NRC staff's review identified the following areas for consideration by the NRC inspection staff during the licensee's implementation of the proposed MELLLA+ (See Section SAR 10.4 for additional detail):

- Steam Separator-Dryer Performance
- Average Power Range Monitor Calibration
- Core Performance
- Pressure Regulator
- Water Level Setpoint Changes
- Neutron Flux Noise Surveillance

These areas are recommended based on the proposed testing for implementation of MELLLA+ at the GGNS site, the extent and unique nature of changes necessary to implement the proposed MELLLA+ domain, and new conditions of operation necessary for operation in the proposed MELLLA+ domain. They do not constitute inspection requirements but are intended to give inspectors insight into important bases for approval of the MELLLA+ LAR.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Mississippi State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been two public comments on such finding published in the *Federal Register* on December 2, 2014 (79 FR 71453). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 PUBLIC COMMENTS

On December 2, 2014, the NRC staff published a "Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed Significant Hazards Consideration Determination, and Opportunity for a Hearing," in the *Federal Register* associated with the proposed amendment request (79 FR 71450). In accordance with the requirements in 10 CFR 50.91, "Notice for public comment: State consultation," the notice provided a 30-day period for public comment on the proposed no significant hazards consideration (NSHC) determination. Public comments were received regarding the proposed amendment (References 65 and 66). Some of the issues discussed in the public comments do not specifically pertain to the proposed NSHC determination. However, the NRC staff has addressed both the issues within the scope of the proposed NSHC and those that are not within the scope. A summary of the comments and the NRC staff responses are provided below.

From Public Commenter No. 1 (Reference 65)

This comment states, in part:

1) In a letter less than a year ago regarding the Monticello Nuclear Reactor which is less than half the size of Grand Gulf it was pointed out: "Because MNGP [Monticello Nuclear Generating Plant] has a small core with low power density, ATWS events with timely operator actions are predicted to cause cladding temperatures well below the regulatory limit... MELLLA+ applications with larger cores and higher power densities may result in instabilities that require the use of heat transfer models in TRACG04 for conditions that are still under NRC review."

NRC Response:

The NRC staff acknowledges that GGNS has a larger core and higher power density compared with Monticello. As such, the licensee has proposed mitigating strategies that will allow the GGNS to operate safely at the MELLLA+ conditions.

The NRC staff has completed a review of TRACG04 model that is used to predict fuel overheating in case of large unstable power oscillations, and its application to GGNS ATWS-I (Instability) events. The main findings from the NRC staff review are:

1. The revisions of GGNS operator training and the associated operator testing of their critical action timing provide reasonable assurance that the reactor water level will be reduced within 90 seconds of ATWS initiation.

2. ATWS-I calculations show that, with prompt water level reduction, large amplitude unstable power oscillations do not have time to develop and there is reasonable assurance that fuel overheating limits are not challenged.

   Thus, the NRC staff concluded that the TRACG calculations performed for ATWS-I for GGNS are sufficient to provide reasonable assurance of compliance with the applicable ATWS regulatory criteria; namely demonstrating that core coolability is maintained during ATWS-I events because T_{min} is not reached.

From Public Commenter No. 2 (Reference 66)

This comment states, in part:

1) It has been shown by nuclear industry experts including both NRC and US' Govt's Brookhaven Lab amongst others that there may be significant hazard posed by allowing Grand Gulf Nuclear Station, Unit 1 (GGNS) to operate in the expanded Maximum Extended Load Line Limit Analysis Plus (MELLLA+) domain if there were to be certain types of problems or failures with different systems. Such possible events include unlikely yet potential 'ATWS' (anticipated transients without scram) caused by certain situations that would cause a few problems including making depressurization necessary (which would cause issues itself). Testing for this specific situation was not completed due to not having the proper codes.

NRC Response:

Section 3.7, "Special Events," of this SE include the NRC staff's evaluation of ATWS. The NRC staff concluded that the licensee has demonstrated that ARI, SLCS and manual FW runback systems have been installed, and they will continue to meet the requirements of 10 CFR 50.62 and the analysis acceptance criteria.

The licensee used "proper" codes as explained in Section 3.4.3, "Analytical Methods," of this SE. This SE includes a description of the TRACG code used for transient analysis. In addition as stated in the SE, "TRACG04 is currently approved for use in DSS-CD and ATWS analysis, and has been used for ATWS best-estimate calculations; however, the licensing basis ATWS analyses are based on ODYN."

This comment states, in part:

2) The problems of Mark III containment, which uses the system at Grand Gulf as the model, are detailed in an NRC document written by Schroeder, J.A., Pafford, D.J., Kelly, D.L., Jones, K.R., & Dallman, F.J. (EG and G Idaho. (1991). An assessment of BWR (boiling water reactor) Mark III containment challenges, failure modes, and potential improvements in performance. doi: 10.2172/6051208 at the following link, and also attached where indicated below: http://www.nrc.gov/reading-rm/doc-collections/insp-manual/temp-instructions/ti-2515_174.doc.

   The current systems used and proposed are not built to accommodate the higher power load being sought for approval, and have not been adequately tested for possible worst case scenarios, which we know are possible.

NRC Response:

The NRC staff has concluded that the GGNS meets all containment safety requirements for operating in the MELLLA+ domain at the EPU power level. These requirements cover the design basis and special events as found in 10 CFR 50. The INEL report referenced in the comment, NUREG/CR-5529, covers severe accident events beyond those required to be addressed in the regulations. The comment on "higher power load," which we assumed meant "higher power density" is addressed above in the response to Commenter No. 1.

9.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

10.0 REFERENCES

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66. Comment (3) of Unknown Individual on Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving Proposed No Significant Hazards Considerations and Containing Sensitive Unclassified Non-Safeguards Information and Order Imposing Procedures for Access to Sensitive Unclassified Non-Safeguards Information (ADAMS Accession No. ML15009A034)

67. Mulligan, K., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Corrected Technical Specifications Pages Regarding Maximum Extended Load Line Limit Plus Amendment Request, Grand Gulf Nuclear Station, Unit 1, Docket No. 50-416, License No. NPF-29," dated August 13, 2015 (ADAMS Accession No. ML15226A074).

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Date: August 31, 2015

Appendices:
A. Request for Additional Information Evaluation
B. List of Acronyms

APPENDIX A

REQUEST FOR ADDITIONAL INFORMATION EVALUATION

This appendix provides a summary of the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the licensee's responses to requests for additional information (RAIs) 1 through 25, documented by letters dated August 26, September 10, and October 2, 2014; and January 20, February 9, and February 19, 2015 (References 7, 9, 10, 11, 17, 18 and 20, respectively) and RAIs R1 and R2, documented by letters dated March 10 and September 4, 2014 (References 3 and 8, respectively).

RAI-1, Power Density > 50 Megawatt Thermal/MLBM/HR

> Section 2.2.1, "Safety Limit Minimum Critical Power Ratio," states that, "The currently approved off-Rated Core flow (CF) uncertainty applied to the Single Loop Operation (SLO) is used for the minimum CF statepoint D and at 55.0% of CF statepoint C." Section 2.2.5 "Power-to-Flow Ratio" states that statepoint C has a power density of 57.42 Megawatts Thermal/Million Pounds/Hour (MWt/Mlbm/hr), which is larger than the MELLLA+ [Maximum Extended Load Line Limit Analysis Plus] Licensed Topical Report (LTR) limit of 50 MWt/Mlbm/hr, and states "this limitation is resolved in the near-term by applying additional conservatism to the cycle-Specific Safety Limit Minimum Critical Power Ratio (SLMCPR)." This "additional conservatism" is not documented in Section 2.2.5 of the Safety Analysis Report (SAR). Provide:
>
> 1. Definition of the "additional conservatism" method.
> 2. A numerical example of the application of this conservatism.
> 3. A justification that the power distribution uncertainties at the higher power density are covered by the proposed method.

Resolution:

In the response, the licensee described the "additional conservatism" as the use of the more conservative SLO flow uncertainties to calculate the SLMCPR. For the first Grand Gulf Nuclear Station, Unit 1 (GGNS) MELLLA+ implementation (Cycle 20), the SLO uncertainty is estimated to be equivalent to a penalty of 0.04 (increase of two-loop operation (TLO) SLMCPR from 1.11 to 1.15), which is a significant penalty.

The licensee provided comparisons of TIP data versus PANACEA calculations that show that, in spite of the large power density, the power distribution uncertainties in GGNS are within acceptable limits. The TIP comparison with calculations indicate that for the typical condition, the power distribution uncertainty is between 3 and 6%, depending on the type of uncertainty (see Table 3 below). Table 3 demonstrates that GGNS is not an outlier plant when compared to the rest of the fleet. Note that these uncertainties are the ones that have been measured during EPU operation in GGNS (up to 44.07 MWt/Mlbm/hr) and their impact on the SLMCPR is already reflected in the current values. The expectation is that an increase from 44 to 50 MWt/Mlbm/hr would increase these uncertainties by a relatively small amount, which should be covered by the 0.02 SLMCPR penalty.

Table 3 Average TIP-PANACEA Uncertainties Showing That GGNS is Not an Outlier

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Based on the above, the NRC staff considers this RAI issue resolved.

<u>RAI-2, Specific Safety Limit Minimum Critical Power Ratio</u>

Section 2.2.1, "Safety Limit Minimum Critical Power Ratio," states that "a +0.02 SLMCPR adder will be added to the cycle-specific SLMCPR."

1. Provide a list of SLMCPR adders in MELLLA+ with respect to Operating Licensed Thermal Power (OLTP) conditions.
2. Specify which adders are part of the Extended Power Uprate (EPU), and which are MELLLA+ specific.

Resolution:

The SLMCPR adders for GGNS were provided.

For pre-MELLLA+ cycles, GGNS used Revision 3 of the Methods LTR (NEDC-33173P-A (Ref. 29) as their licensing basis. Revision 3 Required a 0.02 SLMCPR adder for operation up to EPU conditions, or a 0.03 adder for operation inside the MELLLA+ domain.

Revision 3 of the Methods SER has now been replaced by Revision 4, which is the licensing basis for the GGNS MELLLA+ application. Revision 4 has replaced the Revision 3 SLMCPR adders. Revision 4 adders are based on the power flow ratios, not the region of operation. For operation with power flow ratio lower than 42 MWt/Mlbm/hr, Revision 4 applies a 0.01 SLMCPR adder. For power flow ratios greater than 42 MWt/Mlbm/hr, Revision 4 applies 0.02.

The GGNS MELLLA+ application uses a 0.02 SLMCPR adder for points in the operating map with power/flow ratios greater than 42 MWt/Mlbm/hr. For points with power to flow ratio less than 42 MWt/Mlbm/hr, a 0.01 SLMCPR adder is used. This is an acceptable application of the limitations and conditions of the currently applicable Methods SER (i.e., Revision 4).

Based on the above, the NRC staff considers this RAI issue resolved.

RAI-3, Void Fraction

Figures 2-3, 2-4, and 2-5 of the SAR indicate that GGNS is an outlier with respect to core exit void fraction. GGNS has the highest exit void fraction of all the plants considered, and it approaches ~88% at some points during the cycle.

1. Provide justification about the applicability of General Electric, Hitachi (GEH) methods at this high void fraction.

2. Provide justification that TGBLA06 generates accurate lattice cross sections at void fractions as high as 87%. Please refer to previous approvals that evaluated void fraction levels this high, when applicable.

Resolution:

In its letter dated August 26, 2014, the licensee states, in part that "[t]he response to several RAIs in the Methods LTR...documents the acceptability of exit void fractions greater than 90% and the specific TGBLA06 adequacy of the extrapolation of lattice parameters to in-channel 90 percent void fraction." Based on the above, the NRC staff considers this RAI issue resolved.

RAI-4, Standby Liquid Control System Shutdown Margin

Section 2.3.3, "Standby Liquid Control System Shutdown Margin (SLCS) Shutdown Margin," states that, "The MELLLA+ operating conditions do not change the methods used to evaluate the SLCS shutdown margin."

1. Is SLCS shutdown margin evaluated with all rods out or with a pre-planned rod sequence pattern?

2. Does operation with initial conditions consistent with statepoints C or D (which correspond to the highest rod line) affect the SLCS shutdown margin?

Resolution:

The SLCS shutdown margin is evaluated in a Cold-All-Rods-Out configuration; therefore, operation at statepoints C or D does not affect the answer. The NRC staff considers this RAI issue resolved.

RAI-5, Detect and Suppress Solution-Confirmation Density

1. Have the Backup Stability Protection (BSP) regions been evaluated for the GGNS equilibrium cycle? Provide them if available. If not, where will they be documented? Will they be part of the Supplemental Reload Licensing Report (SRLR)?

2. Describe the criteria used to set the Oscillation Power Range Monitor (OPRM) armed region.

3. Provide justification that the OPRM armed region defined as 75% drive flow shown in Figure 2-18 is conservative for GGNS MELLLA+ cooperation.

Resolution:

The BSP regions were provided. The 75 percent drive flow is the generic Detect and Suppress Solution-Confirmation Density (DSS-CD) armed region. The NRC staff considers this RAI issue resolved.

RAI-6, Increased Moisture Carry-Over

Section 3.3.4, "Steam Line Moisture Performance Specification," states that, "The highest Moisture Carryover (MCO) predicted under MELLLA+ conditions is less than 0.2 wt %" ... "The amount of time GGNS is operated with higher than the original design moisture content (0.10 wt %) is minimized by operations" ... "The maximum permissible MCO leaving the Reactor Pressure Vessel (RPV), above which Mainsteam Line (MSL) components could begin to degrade as a result of the high moisture content in the steam, was found to be 0.33 wt %."

Provide a summary explanation of:

1. What analyses were performed to determine the 0.33% permissible limit?

2. What analyses were performed to determine the 0.2% MCO under MELLLA+ conditions?

3. What plant operations are used in GGNS to minimize the MCO?

4. Provide a short physical explanation of what causes the increased MCO at lower flow. Is this mechanism predicted using an experimental correlation or a first principle analytical tool?

5. How is the MCO monitored during operation? What is the typical surveillance period?

Resolution:

A detailed explanation was provided describing the reasons for the increased MCO (lower flow rate through the separators, which reduce their efficiency because of reduced centrifugal forces) and the consequences (mainly increased transport of contaminants to the cleanup systems). The NRC staff reviewed the consequences of the increased MCO in Sections 3.3.3, 8.5.1, 8.6.1, and 9.2.1.4 of the SE. Based on these reviews, the NRC staff considers this RAI issue resolved.

RAI-7, Reactor Core Isolation Cooling Net Positive Suction Head

Section 3.9.3, "Reactor Core Isolation Cooling (RCIC) Net Positive Suction Head (NPSH)," states that, "The RCIC system has the capability of using the Condensate Storage Tank (CST) or the Suppression Pool (SP) as a suction source" ... "GGNS calculations demonstrate that the RCIC pump would have adequate NPSH and low suction pressure trip margins given a SP water temperature of 140 °F".

1. Is the CST available for RCIC even under containment isolation conditions?

2. If the SP temperature reaches >140 °F, what indication/training does the operator have to switch from SP to CST inlet?

Resolution:

The CST is available for RCIC under containment isolation conditions, and it is the preferred injection source. The operator indicators were provided. The NRC staff considers this RAI issue resolved.

RAI-8, Large Break and Small Break LOCA

Section 4.3.1, "Break Spectrum Response and Limiting Single Failure," states that, "A number of small break sizes were evaluated at the rated Current Licensed Thermal Power (CLTP)/Rated Core Flow (RCF)."

1. Provide a list of cases evaluated and indicate the limiting case.

2. The evaluation was performed at RCF. Provide an explanation why the results will not be significantly different at minimum (80%) or maximum (105%) core flow.

3. The Small-Break Loss-Of-Coolant Accident (SBLOCA) results in Table 4-4 show the top peaked axial power shape is limiting compared to the mid-peaked power shape. The results for Large-Break Loss-Of-Coolant Accident (LBLOCA) in Table 4-3 show the mid-peaked axial power shape being limiting. Explain the difference in these results.

4. In Section 4.3.2, explain the following regarding Table 4-3:

   a. Why the mid-peaked axial power shape is analyzed and a calculation for the top-peaked or bottom-peaked axial power shape is not shown.

   b. Why the mid-peaked axial power shape is limiting in terms of the Peak Cladding Temperature (PCT) difference between the value of first peak at mid-peak and the value at top-peaked axial power shape.

   c. Why the first peak is lower than the second peak for the mid-peaked axial power shape calculation at 100% power MELLLA+ condition and the second peak is higher than first peak at the Appendix K condition.

   d. Please provide a plot of PCT versus time for the LBLOCA top- and mid-peaked axial power shape cases.

Resolution:

The requested information was provided. A table with small break size results was provided, and the limiting case is a recirculation suction line break with an area of 0.08 ft^2, and justification is provided for calculating these breaks only at nominal flow. A study was provided to evaluate the impact of top- vs middle- vs top-peaked power shapes for small and large breaks. All PCT results are acceptable. Based on the above, the NRC staff considers this RAI issue resolved.

RAI-9, Anticipated Operational Occurrence (AOO) Impact of Flow

On a separate MELLLA+ submittal (GEH Report Spec: 000N2436, dated 1/12/2014), data was provided to justify that Anticipated Operational Occurrences (AOOs) have smaller ΔMCPR [delta minimum critical power ratio] at 80% core flow than at 105%. However, in Table 9.1 of the SAR, most AOOs, but not the limiting one, have a larger ΔMCPR at 80% flow than at 105%. The argument presented in the past is a shift in power towards the bottom as the voids increase for then 80% flow case, which results in increased control rod performance.

1. Provide the initial axial power shapes for the events in Table 9-1 at 80% and 105% flow.

2. For all cases in Table 9.1, the transient peak power is lower at 80% than at 105% flow yet the ΔMCPR is larger. This is counterintuitive. Please provide an explanation.

Resolution:

The information was provided. The probable cause of the unusual ΔMCPR behavior is likely to be the reduced critical power ratio (CPR) performance of GNF2 fuel at low flows. The NRC staff considers this RAI issue resolved.

RAI-10, Bi-Stable Flow

Is GGNS susceptible to bi-stable flow in the recirculation loops? If so, what is the maximum achievable recirculation flow used in normal operation to minimize bi-stable flow concerns?

Resolution:

The information was provided. Bi-stable flow oscillations are present at some flow conditions in GGNS, but they do not affect the maximum attainable flow, which is ~102.5 percent. The NRC staff considers this RAI issue resolved.

RAI 11, Plant Design Parameters

1. Provide plant design parameters relevant to the Anticipated Transient Without Scram (ATWS) calculations in Section 9 of the SAR. Specifically: turbine bypass capacity, sources of high pressure injection and their operability issues (e.g., steam is lost after isolation ...), sources of low pressure injection and their operability issues (e.g. CST pumps ...).

2. Provide vessel component elevations in units comparable to the ones used for water level in the Section 9 figures (include separators, Feedwater (FW) spargers, nominal level, level set points for actuations, Top of Active fuel (TAF) ...).

3. What is "ATWS water level" in Figure 9-8 of the SAR?

Resolution:

The information was provided in forms of tables with requested data. The NRC staff considers this RAI issue resolved.

RAI-12, ATWS Sequence of Events

Provide tables of the assumed sequence of events for the ODYN licensing calculation, the ATWS best estimate calculation, and the ATWS/Stability calculation.

Resolution:

The information was provided. The provided tables document the sequence of events for the licensing ATWS calculations and the best estimated ATWS-I event analyzed. The NRC staff considers this RAI issue resolved.

RAI-13, ATWS Calculations

1. Table 9-5 specifies a Boron SLCS concentration of 269%. Please, describe the units (i.e., percent of what?).

2. The SLCS initiation time has been increased from 120 seconds at CLTP to 300 seconds at MELLLA+. Table 9-6 specifies that this increase is the main reason why the calculated ultimate suppression pool temperature increases significantly (165.3 °F vs 197.5 °F). Is there a specific reason for the increase? Is the 300 seconds consistent with operator actions in the simulator?

3. The Licensing Basis ODYN ATWS Analysis calculates a suppression pool temperature of 197.5 °F. Are NPSH limits satisfied by all the equipment assumed operable by the ODYN calculation? If a transfer to CST as the source of Emergency Core Cooling System (ECCS) cooling is assumed, provide the timing of the transfer.

Resolution:

The information was provided. CST is available as the primary source of coolant as long as certain conditions are met (primarily suppression pool maximum level is not exceeded). The SLCS initiation time was increased to 300 second for consistency and to represent expected operator actions. The NRC staff considers this RAI issue resolved.

RAI-14, Safety Relief Valve Setpoints and Out-of-Service Allowance

Table 9-5 specifies that the ATWS transient was run with 20 total Safety Relief Valves (SRVs) with five SRVs out of service for both CLTP and MELLLA+, and Section 9.3.1.1 states "With the safety function of at least nine SRVs and the relief function of at least six SRVs operable".

1. Describe the difference between "safety function" and "relief function." Describe why the safety valve flow in Figure 9-1 is ~25% of the relief valve flow. How many valves are open in the case of Figure 9-1?

2. Has the number of allowed SRVs out of service been changed as a result of the MELLLA+ operating domain extension?

3. The SRV Analytical Opening Setpoint in Table 9-5 has been increased from 1,183 to 1,246 psig, which is greater than the 3% drift tolerance described in the text? Please elaborate and justify the change.

Resolution:

The information was provided. GGNS is a boiling-water reactor (BWR)/6, which was built with a significant excess SRV capacity. This is the reason why five SRVs can be maintained out of service. The number of SRVs out of service was decreased from seven to five during the EPU licensing, which increased the power to 115 percent. The MELLLA+ upgrade does not change the core power or steam flow significantly and does not require additional SRV capacity. The NRC staff considers this RAI issue resolved.

RAI-15, ATWS Water Level Strategy

Section 9.3.1.1, "ATWS (Licensing Basis)," specifies that, "Water level control per procedures." Section 9.3.1.2 "ATWS (Best-Estimate Calculation)" specifies

"Water level control using the designated water level control strategy." Section 9.3.3 "ATWS with Core Instability" states that "Reactor water level was controlled at approximately top of the active fuel (TAF) after a 90-second delay following indication of no scram."

1. Provide a detailed description of what water level control strategy (with emphasis on timing) was used for each calculation.

2. Describe the sources of water used to control the level. If FW pumps were used, describe automated actions (i.e., loss of extraction steam), and assumptions about operability (i.e., residual steam volume, if used) after the Main Steam Isolation Valve (MSIV) isolation occurs.

3. The best estimate ATWS calculations (Figures 9-8 and 9-10) show some degree of high-pressure injection before time ~500 sec. Is this injection consistent with the available plant equipment?

4. Has the water level control strategy and timing changed as a result of the MELLLA+ domain extension?

5. Are there any operator training concerns/changes as part of the MELLLA+ domain extension and the 90-second delay?

6. Have the 90-second water level control and 40-second depressurization delays been tested in the plant simulator?

7. Figures 9-8 and 9-10 appear to show flow injection (red line) for times between 500-1000 during the depressurization. Do GGNS Emergency Operator Procedures (EOPs) require the termination of all high pressure flow injection except SLCS during the depressurization phase? Is the calculation consistent with EOPs?

Resolution:

The information was provided. The water level strategy during ATWS has not changed for MELLLA+ and remains to reduce the water level to ~TAF (within a band of approximately 2 ft.). Water level is allowed to go into the core region to the minimum steam cooling water level (MSCWL) if required. The strategy was demonstrated to the NRC staff during audit simulator demonstrations (Reference 64). Based on the above, the NRC staff considers this RAI issue resolved.

RAI-16, Pressure Control Strategy

Operators may choose to perform a controlled partial depressurization to: (1) obtain a larger Heat Capacity Temperature Limit (HCTL) margin and avoid emergency depressurization, and (2) allow the use of mid-to-low pressure injection sources like the CST pumps. Have operator actions in the simulator and training been reviewed to ensure that the licensing ATWS calculations are conservative?

Resolution:

The information was provided. The calculations are consistent with simulator operator actions. The partial depressurization strategy to maintain margin to HCTL was demonstrated to the NRC staff during an audit (Reference 64). The NRC staff considers this RAI issue resolved.

RAI-17, Boron Mixing and Transport

Figure 9-8 shows the boron reactivity stabilizing at ~500 seconds, then increasing at ~1000 seconds followed by a significant decrease. However, Figure 9-10 shows a significant decrease in boron reactivity at ~1000 seconds. Please explain the phenomena that lead to such significantly different behavior.

Resolution:

The information was provided. The phenomena are caused by some transient void conditions caused by the control system attempting to control pressure at 50 pounds per square inch gauge (psig). The NRC staff considers this RAI issue resolved.

RAI-18, Detailed Plots

The plots provided in Section 9 are difficult to read.

1. Provide enhanced neutron flux plots, where the axis is limited to 100% CLTP for all best estimate ATWS calculations.

2. The neutron flux provided for the ATWS-Instability (ATWS-I) calculation is core-average. Provide additional plots with hot channel powers at symmetric core locations showing the amplitude of the regional oscillations for the ATWS-I calculation.

Resolution:

The information was provided. Detailed plots show the hot channels oscillating out of phase. The NRC staff considers this RAI issue resolved.

RAI-19, PCT

1. The ATWS plots of PCT (Figures 9-9 and 9-11) show two distinct PCT heat up ramps. One occurs early in the transient, and a second one occurs at ~500 sec in Figure 9-9 and 9-11 when depressurization starts, with a period of low temperature in between. Are the hot rods in dryout condition during the heat-up ramps? Describe what phenomena causes the rewetting (low temperature) at ~500 sec.

2. The ATWS-I calculation shows a PCT heat up at ~80 sec when the power oscillations initiate. The PCT recovers and the rods seem to rewet at ~130 sec when the oscillations are mitigated by the flow reduction. What mechanism allows for heatup and rewet?

3. Provide plots similar to Figures 9-9, 9-11, and 9-13 that shows PCT superimposed with the calculated minimum stable film boiling temperature (T_{min}) value.

4. Provide plots showing the calculated margin between PCT and T_{min} for Figures 9-9, 9-11, and 9-13.

Resolution:

The information was provided. Even though T_{min} is not reached during this transient, critical heat flux (CHF) is, resulting in a decrease of heat transfer capability that causes the temperature increase. When CHF conditions are recovered, normal nucleate boiling is established, increasing the heat transfer capability. The NRC staff considers this RAI issue resolved.

RAI-20, Code-to-Code Comparison

Events leading to reactor instabilities cause oscillations in PCT over time. The magnitude of these oscillations has been seen to vary from code to code. Analyses completed by the Office of Nuclear Regulatory Research at the NRC have documented TRACE results for ATWS-I that lead to reactor instabilities with high PCT.

1. Develop a synonymous model using TRACG.

2. Compare TRACG results with TRACE results for an ATWS-I turbine trip with 100% bypass event initiated from 120% Originally Licensed Thermal Power (OLTP) and 85% reactor core flow at the beginning of cycle and the peak hot excess point in the cycle. Provide discussion of differences between the two calculation results, in particular, wherever possible,

identify candidate constitutive models, modeling procedures, input assumptions, or other factors that contribute to the differences.

3. Provide results in tabular form and in plots of the same two cases in RAI 20.2 above (ATWS-I turbine trip with 100% bypass event initiated from 120% Originally Licensed Thermal Power (OLTP) and 85% reactor core flow at beginning of cycle and the peak hot excess point in the cycle) using a constant T_{min} of 900K.

<u>Resolution</u>:

The NRC staff provided a TRACE-PARCS model of a generic BWR to the licensee. A synonymous TRACG04 model was generated, and care was taken to make the two models as similar as possible. However, it must be recognized than with two different codes and a complete plant model of this complexity, differences between the TRACE and TRACG04 models are unavoidable. Table 20-1 of the RAI response attempts to summarize the characteristics of both code models.

To avoid problems associated with the use of the different T_{min} correlations in TRACE and TRACG04, fixed values of T_{min} were used at 725K and 900K for the comparisons. Cases were run for Peak Hot Exposure (PHE) and beginning of cycle (BOC). As expected, the oscillations were more pronounced on both codes for the PHE case.

For all cases run, after tripping the recirculation pumps, TRACE predicted slightly lower natural circulation flow than TRAG04, even though the water levels and FW temperature were similar for both codes. Unstable power oscillations occurred earlier in TRACG04 and were of larger magnitude. Figures 20-28 and 20-29 from the RAI response show the main results for a high T_{min} value of 900K at BOC conditions.

Figure 20-28 Reactor Power, Tmin = 900K, BOC

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Figure 20-29 Core Flow, Tmin=900K, BOC

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Note: The cause of the flow difference between TRACE and TRACG04 has not been investigated in detail by the NRC staff, but it can be noted that the flows do not really agree for any time, including t>100 s (see Figure 20-29), because the powers are completely different for t>100 s (see Figure 20-28) and natural circulation flows should be significantly different. Several hypothesis have been proposed (but have not been analyzed in detail) including differences in the separator modelling and/or the channel inlet geometry. Nevertheless, positive conclusions can be drawn from this code-to-code comparison.

Both codes predict the same PCT behavior. Figure 20-0 from the RAI response summarizes the PCT conclusions. For a low T_{min} value (725K), once the oscillations develop large enough to reach T_{min}, the fuel cladding overheats rapidly and both codes predict failure of coolable geometry (T > 2200 °F). When the T_{min} value is assumed large (900K), both codes predict the oscillation will never reach T_{min} and the clad temperature behavior is similar, and it maintains coolable geometry.

Figure 20-0 PCT Comparison, PHE

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The next figure shows a detail of the fast PCT overheating when T_{min} is reached in both codes. As seen in the figure, the PCT starts showing periodic dry out and rewet when oscillations become large enough to challenge the CPR limit. As long as the PCT oscillation does not reach T_{min}, there is rewetting when the power/flow oscillation enters a regime bellow CHF. On the next oscillation, CHF is reached, and PCT overheating starts. This dry out-rewet cycle continues with the period of the power oscillations (~2 sec). When the oscillations become large enough so that Tmin is reached, even though CHF conditions are no longer satisfied, the codes do not allow rewetting and the PCT excursion ramps up to a very high value so that core coolability is compromised.

Figure 20-4. Maximum Peak Clad Temperature, $T_{min} \approx 725$ K, PHE

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Based on the above, the NRC staff considers this RAI issue resolved.

RAI-21, Steam Dryer Structural Integrity

1. Are the moisture carryover values or steam quality for steam (a) entering the steam separator, (b) exiting the steam separator, (c) entering the steam dryer, and (d) exiting the steam dryer affected by MELLLA+ core flow conditions?

2. Are the boundary conditions used in Acoustic Plant Based Load Evaluation (PBLE) model affected by MELLLA+ flow? Is there any impact on reactor water level & boundary conditions for annular region between dryer skirt and separator stand pipes; and annual region between RPV wall and dryer skirt? Is there any impact on dryer pressure loading used and on dryer structural analysis?

3. Are the stresses in the steam dryer evaluated for EPU conditions bounding for plant operation at EPU conditions combined with MELLLA+ conditions?

Resolution:

The licensee provided a response to RAI-21 by letter dated August 26, 2014 (Reference 7), which included a summary of its evaluations. The NRC staff reviewed the licensee's steam dryer evaluations for the combined MELLLA+ and EPU conditions and found them acceptable. The stresses in the steam dryer meet the acceptance criteria for the flow-included vibration (FIV) stress limit (13600 psi endurance limit for high cycle fatigue) (Reference 35).

The NRC staff concludes that the proposed license amendment to operate GGNS at the proposed EPU conditions, combined with MELLLA+ for the steam dryer, is acceptable with respect to potential adverse flow effects for high cycle fatigue, as well as the ability to withstand the ASME normal, upset, emergency, and faulted load combinations. The NRC staff also concludes that the steam dryer will maintain its structural integrity for the combined EPU and MELLLA+ flow conditions. This steam dryer is discussed in greater detail in SE Section 3.3, GGNS SAR Sections 3.3.3, "Steam Separator and Dryer Performance," and 3.3.4, "Steam Line Moisture Performance Specification." Based on the above, the NRC staff considers this RAI issue resolved.

RAI-22, Core Design

1. The SRLR will validate that the power distribution in the core is achieved while maintaining individual fuel bundles within the allowable limits as defined in the Core Operating limits Report (COLR). When the SRLR and the COLR will be available for GGNS MELLLA+ operation?

2. Provide the details to obtain a final loading pattern including procedure, guidance, criteria, and approved methodologies used for this analysis in relation to GESTAR II.

3. Table 2-1 and Figures 2-1 through 2-6 indicate the core design and fuel monitoring parameters for each exposure statepoint. Table 2-1 shows the peak nodal exposures starting from 38.849 to 56.660 GWd/ST [gigawatt days per short ton] (54.272 GWd/ST for GGNS M+ SAR at equilibrium 115% OLTP) and Figure 2-1 through 2-6 only shows cycle exposure up to 18 GWd/ST.

    a. Why do the figures only show the data up to 18 GWd/ST?

    b. Provide values for maximum bundle power, flow for peak bundle, exit void fraction for peak power bundle, maximum channel exit void fraction, core average exit void fraction, and peak linear heat generation rate (LHGR) at peak nodal exposure.

    c. Why isn't the peak nodal exposure data for GGNS M+ at equilibrium – 120% OLTP included in Table 2-1?

4. Provide core maps to show the bundles that experienced the 0.1% boiling transition criterion.

5. Provide a detailed description of the GGNS MELLLA+ core design in response to the core instability and fuel bundles that experienced boiling transition. Include any relationship among hot channels, regional instability experienced in Figure 9-13, and core loading pattern.

6. Since the SRLR is not ready at this moment, please provide a detailed description and basis that the operational conditions for GGNS in the MELLLA+ operating domain are within expected parameters based on the data shown in Figures 2-7 through 2-15.

Resolution:

The information was provided. The SRLR for Cycle 20 was provided to the NRC staff, and the requested data was verified. The NRC staff concludes this RAI issue resolved.

RAI-23, Technical Specifications

1. Please provide justification for changing the pump discharge pressure of the SLCS from 1340 to 1370 psig in Surveillance Requirement 3.1.7.7.

2. Provide approved methodologies used to support the proposed addition of TS 5.6.5.a.6.

Resolution:

The information was provided. The new requirement is a consequence of the higher peak pressure calculated for ATWS events under MELLLA+. The SLCS piping is designed to operate up to 1700 psi, and the relief valve is set at 1700 psi. The NRC staff considers this RAI issue is resolved.

RAI-24, Turbine Trip Events

Results for TTNBP [Turbine Trip Without Bypass] during an AWTS-I event are not included in the SAR. Provide results for TTNBP in the MELLLA+ operating domain.

Resolution:

The information was provided. The plots were provided in the RAI response. The NRC staff considers this RAI issue resolved.

RAI-25, Simulator Update

1. Describe up-to-date training status of the key operator actions credited in the TRACG ATWS instability analysis.
2. Provide the schedule when the GGNS Simulator will be completely updated for operators' training in the MELLLA+ operating domain.

Resolution:

By letter dated February 19, 2015 (Reference 20), the licensee proposed the following license condition to address the operator training for key operator actions. The NRC staff conclude that this license condition provides the verification and validation of the training needed to support operation in the MELLLA+ domain. See SE Section 4.1 for the license condition regarding operator training. Based on the above, the NRC staff considers this RAI issue resolved.

RAI-R1

By letter dated September 4, 2014 (Reference 8), RAI 1 states

> Section 9.3.3 of Attachment 4 states that zirconium credit will be used in the Shumway correlation. The NRC staff has reviewed the zirconium data described in letter dated September 9, 2013 (ADAMS Accession No. ML13253A105), from GE Hitachi Nuclear Energy and has the following questions:
>
> a. Fresh fuel that has little or no oxide at the start of the transient and the fresh fuel is often in the more reactive part of the core; therefore, it is the data of interest. Figure 6 of the September 9, 2013, letter has many data sets provided together on it. Provide a plot showing clean zirconium separate from all the other data including SS [stainless steel], Inconel, zirconium oxide and any data that would be expected to have oxidized zirconium.
>
> b. In the September 9, 2013 letter, much of the Hoffman (FKZ) data that has a rapid cooling rate early in the experiment that may have been incorrectly interpreted as quench. A closer examination of the data shows this initial cooling was due to startup of the test and that quench occurred much later at significantly lower temperatures. In addition, because of high temperatures during the tests and pre-conditioning of the rods, the rods are likely have thick oxide layer thicknesses that are not representative of fuel in a commercial reactor undergoing an Anticipated Transient without Scram with instability (ATWS-I) event. Provide a plot

showing the data considered valid by the licensee for justifying Shumway as implemented in TRACG for the intended application (e.g., ATWS-I under MELLLA+). Textual justification to support the choice of the selected data and exclusion of the other data should be provided focusing on comparing the applicable test conditions to the plant application conditions to support a conclusion that the data is applicable.

<u>Resolution</u>:

The response to the RAI 1 provides the licensee's justification for the zirconium credit used in the Shumway correlation. In Part (a) of the RAI, the NRC staff requested that the licensee address that fresh fuel may have little or no oxide layer. The staff notes that taking credit for zirconium oxide properties in the Shumway correlation would result in a substantial increase in the T_{min} predicted by the correlation. In the response the licensee states that the limiting fuel bundles would likely be the once-burnt fuel bundles. While this is an accurate description of the peak reactivity exposure for many modern fuel bundle designs that incorporate burnable gadolinia bearing fuel rods, this does not address that fresh fuel may be more limiting than partially burnt fuel.

First, since typical operation of a BWR results in lower-peaked axial power distributions near the BOC, and that a lower-peaked axial power shape tends to be less stable, it cannot be ruled out that the limiting point in cycle would be BOC.

Second, during ATWSI events for MELLLA+ plants, the reactor can be expected to become highly unstable. In large reactor cores, higher harmonic modes of the neutron flux may become unstable. The non-linear coupling and complexity of the modal dynamics may result in highly complex neutron flux oscillation contours. It may be possible for higher than first-order harmonics to become excited; and further, depending on the control rod pattern, the first harmonic may be a radial as opposed to an azimuthal mode. Given this complex nature of the dynamics of the power distribution during the instability, it is not possible to categorically determine a priori that the most limiting bundle within the reactor core is one that exhibits a higher bundle reactivity or a higher power during steady-state operation.

Based on the description of the implementation of the Shumway correlation in TRACG (Reference 62) and the NRC staff's audit of the TRACG code relating to T_{min} (Reference 60), TRACG does not take into account the local fuel exposure when diagnosing whether a particular node is coated with an oxide layer. Given that (1) the code does not consider whether a fuel element has accumulated an oxide layer, (2) the limiting location may be a fresh fuel bundle without a zirconium oxide layer, and (3) crediting the material properties of zirconium oxide in the correlation produces a higher prediction of T_{min}, the NRC staff concludes that if TRACG was to take credit for the zirconium oxide properties in the Shumway correlation, the results would be potentially non-conservative. The response states that no credit is taken for zirconium oxide in the implementation, and the NRC staff finds that this is reasonable.

The response then provides a discussion of experimental data used to qualify the Shumway correlation for zircaloy. The response references three data sets: (1) Hofmann, (2) Peterson and Bajorek, and (3) Halden.

As to the first data set, the Hofmann data, the NRC staff has reviewed the source documentation of these experimental results and found that some quenching data were questionable based on the likelihood of phenomena such as steam pre-cooling to explain the thermo-couple indications. The response describes how the licensee agrees with the NRC staff's determination in several instances. The staff has reviewed the temperature traces described in Table 1-2 and agrees with the licensee that these traces are applicable. During an audit conducted by the staff at GEH (Reference 60) the criteria for selecting these traces were discussed in greater detail. The NRC staff's audit report provides a more detailed description of the process for selecting acceptable temperature traces.

The Hofmann data, however, only includes useful information for experiments with relatively thick oxide layers (100 or 300 micrometer (um)). The presence of the oxide layer is expected to increase the T_{min} compared to clean zircaloy surfaces. The response estimates the magnitude of this effect to be 50 to 80 K based on a paper by Dhir. At low pressure the data can be compared with the Peterson and Bajorek data, which is collected for zircaloy without oxide. The magnitude inferred by Dhir appears to be roughly consistent with the differences observed in data as reported in Figure 1-1. The NRC staff notes here that the T_{min} measured for very thick (300 um) and thick oxide layers (100 um) appears to have only a minor, if any, effect on measured T_{min}. The staff does not consider this to imply that there is no affect at smaller thicknesses. This data may indicate a "saturation" of the effect of the oxide layer. Therefore, this data should not be used to infer that data collected at any oxide layer thickness would be applicable.

The response provides some justification for the degree of impact based on the methodology of Wendelstorf. The discussion, however, appears to focus on how the presence of the oxide layer has only a minimal impact on the fuel rod thermal resistance; and therefore, only a small impact on the transfer of energy from the pellet to the coolant. The NRC staff did not consider this information in the review as it is not clear how the transfer of energy through the cladding due to thermal resistance and temperature difference would impact the phenomena of liquid film adhering to the cladding surface. The staff's review, therefore, does not endorse the use of this method to estimate the impact of cladding surface oxide layer thickness of minimum stable film boiling temperature.

When considered together, the Peterson and Bajorek data and the Hofmann data indicate that the Shumway correlation appears to reasonably predict the value of T_{min} at lower pressure and low flow conditions. This is reflected in Figure 1-1 of the response. The slightly higher values of the Hofmann data points may be attributable to the thick oxide layer, and the NRC staff agrees that this explanation is reasonable based on the Dhir observations.

Further, the NRC staff agrees with the response in principle in that the amount by which a zirconium oxide layer would reduce the T_{min} has not been well established. The NRC staff finds that the clean zirconium Shumway correlation produces reasonable predictions of T_{min} based on the Hofmann data and the Peterson and Bajorek data.

The higher pressure tests performed at Halden are more representative of ATWSI conditions, and on this point the NRC staff agrees with the response. Figure 1-1 provides a comparison of quench data from the Halden experiments with the Shunway correlation. However, there is some distortion in this comparison. The Shumway correlation includes a term that accounts for the flow (through a Reynold's number dependence). As the Halden tests were conducted with a relatively high flow rate, as discussed in the response, Figure 1-1 is misleading. The Shumway curve is presented at zero flow, but would predict a higher value if the Halden flow rate was explicitly considered in the evaluation of the predicted T_{min}. As such, the degree by which the Halden data points exceed the Shumway curve in Figure 1-1 is not an accurate representation of the conservatism in the correlation. This was reviewed in detail by the staff during an audit conducted at GEH (Reference 60). The impact of the Reynold's number dependence is such that the measured data from Halden and the prediction are in closer agreement with the measurement still exceeding (slightly) the prediction.

Further, there is the matter of oxide accumulation during the Halden tests. GEH estimated the accumulation of oxide during these tests and the NRC staff audited this information (Reference 60). In the review of that information, the staff concluded that the oxide accumulation during the Halden tests is likely to be small enough to have only a small impact on minimum stable film boiling temperature. See Reference 60 for additional details, however, the staff finds the use of the Halden data acceptable for qualification of the Shumway correlation and finds that the correlation produces reasonable predictions for T_{min}. Recall that the Figure 1-1 does not account for the flow rate in depicting the comparison to the Halden data.

Lastly, the NRC staff concurs that the void dependence of the Shumway correlation is not well supported by experimental data. Therefore, the staff agrees with the approach discussed in the licensee's response that the void term is not credited.

On the basis of the experimental qualification, and the current understanding of the phenomena presented in the available literature; the NRC staff finds that the Shumway correlation as implemented in TRACG (clean zirconium properties and no void credit) produces reasonable predictions of T_{min} at low and high pressure.

While the correlation may be conservative, depending on the effects of oxide, subcooling, void fraction, measurement biases, surface roughness, or other effects; the NRC staff does not find that there is sufficient justification to conclude that the current implementation is definitely conservative. Rather, the staff concludes that given the current qualification information presented, the correlation appears to predict the T_{min} with reasonable agreement.

RAI-2

By letter dated March 10, 2014 (Reference 3), RAI 2 states:

> Section 9.3.3 of Attachment 10 states:
>
> > The plant-specific ATWS stability calculation was performed using the NRC approved neutronic and thermal-hydraulic codes ... TRACG04.
>
> An error in the code was corrected by letter from GE Hitachi Nuclear Energy to the staff dated October 15, 2013 titled "Updated TRACG Quench Front Model Description and Qualification", which is currently under staff review. In the October 15, 2013 letter, GE noted:
>
> > The quench front model has been updated to correct an error in the quench front heat transfer coefficient for bottom reflooding, and to better capture the heat transfer ahead of and behind the quench front.
>
> Please clarify which quench model is being used in the application.

Resolution:

The NRC staff conducted an audit for the TRACG code to confirm that the corrections made to the quench model were acceptable (Reference 60). The staff found that the changes were acceptable. Therefore, the response is acceptable.

APPENDIX B

LIST OF ACRONYMS

2RPT	Two-Pump Recirculation Pump Trip
ΔCPR	Delta Critical Power Ratio
ABSP	Automated Backup Stability Protection
AC	Alternating Current
ADAMS	Agencywide Documents Access and Management System
ADS	Automatic Depressurization System
ANSI	American National Standards Institute
AOO	Anticipated operational occurrences
AOPs	Abnormal Operating Procedures
AOR	Analysis of Record
APRM	Average Power Range Monitor
ARI	Alternative Rod Injection
ASME	American Society of Mechanical Engineers
ASME Code	ASME Boiler and Vessel Pressure Code
ATWS	Anticipated Transient Without Scram
ATWSI	Anticipated Transient Without Scram with Instability
AV	Air Volume
BOC	Beginning of Cycle
BOP	Balance of Plant
BSP	Backup Stability Protection
BSW	Biological Shield Wall
BWR	Boiling-Water Reactor
BWRVIP	Boiling Water Reactor Vessel Internals Project
CCF	Common-Cause Failure
CDA	Confirmation Density Algorithm
CDF	Core Damage Frequency
CF	Core Flow
CFFF	Condensate Full Flow Filtration
CFR	*Code of Federal Regulations*
CHF	Critical Heat Flux
CLOD	Current Licensed Operating Domain
CLTP	Current Licensed Thermal Power
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CTP	Current Thermal Power
CS	Core Spray
CSAU	Code Scaling Applicability and Uncertainty

D-3	Defense-in-Depth and Diversity
D&S	Detect and Suppress
DBA	Design Basis Accident
DC	Direct Current
DSS-CD	Detect and Suppress Confirmation Density
DW	Drywell
ECCS	Emergency Core Cooling System
EOPs	Emergency Operating Procedures
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
EQ	Environmental Qualification
°F	Fahrenheit
FAC	Flow-Accelerated Corrosion
FHA	Fuel-Handling Accident
FIV	Flow-Induced Vibration
FMCPR	Final Minimum Critical Power Ratio
FPCC	Fuel Pool Cooling and Cleanup System
FPCS	Fuel Pool Cooling System
FW	Feedwater
FWCF	Feedwater Controller Failure
FWHOOS	Feedwater Heater Out of Service
Gd	Gadolinium
GDC	General Design Criterion/Criteria
GE	General Electric
GEH	General Electric Hitachi
GGNS	Grand Gulf Nuclear Station
GL	Generic Letter
GNF	Global Nuclear Fuel
gpm	gallons per minute
GWd/ST	Gigawatt Days per Short Ton
HCTL	Heat Capacity Temperature Limit
HCU	Hydraulic Control Unit
HEPA	High Efficiency Particulate Air
HPCS	High Pressure Core Spray
HPSP	High Power Setpoint
HSBW	Hot Shutdown Boron Weight
HSI	Human-System Interfaces
HVAC	Heating, Ventilating, and Air Conditioning
HWC	High Water Chemistry
IASCC	Irradiation Assisted Stress-Corrosion Cracking
ICF	Increased Core Flow
ICPR	Initial Critical Power Ratio
IGSCC	Intergranular Stress Corrosion Cracking
IMCPR	Initial Minimum Critical Power Ratio

IRM	Intermediate Range Monitor
LCO	Limiting Condition for Operation
LAR	License Amendment Request
LBLOCA	Large-Break Loss-of-Coolant Accident
LCS	Leakage Control System
LERF	Large Early-Release Frequency
LFWH	Loss of Feedwater Heater
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LTA	Lead Test Assemblies
LRNBP	Load Rejection With No Bypass
LTR	Licensing Topical Report
LTS	(Stability) Long-Term Solution
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCO	Moisture Carry-Over
MCPR	Minimum Critical Power Ratio
M+	(Short for MELLLA+)
MELB	Moderate Energy Line Break
MELLLA	Maximum Extended Load Line Limit Analysis
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
Mlbm/hr	million pounds/hour
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSLB	Main Steam Line Break
MWt	Megawatts Thermal
N-16	Nitrogen-16
NMS	Neutron Monitoring System
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
NTS	Nominal Trip Setpoint
NUMAC	Nuclear Measurement Analysis and Control
OER	Operating Experience Review
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
OPRM	Oscillation Power Range Monitor
ORNL	Oak Ridge National Laboratory
P	Power
P_a	Calculated Peak Containment Internal Pressure
PCT	Peak Clad Temperature

PHE	Peak Hot Exposure
ppm	Parts per Million
PRA	Probabilistic Risk Analysis
PRNM	Power Range Neutron Monitoring
psi	Pounds per Square Inch
psig	Pounds per Square Inch Gauge
Pu	Plutonium
RAI	Request for Additional Information
RCF	Rated Core Flow
RCIC	Reactor Core Isolation Cooling
RCIS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
REM	Roentgen Equivalent Man
RG	Regulatory Guide
RHR	Residual Heat Removal
RIPD	Reactor Internal Pressure Difference
RPC	Rod Pattern Controller
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RRS	Reactor Recirculation System
RS	Review Standard
RSLB	Recirculation Suction Line Break
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
RWL	Rod Withdrawal Limiter
S_{AD}	Amplitude Discriminator Setpoint
SAFDL	Specified Acceptable Fuel Design Limit
SAG	Severe Accident Guidelines
SAR	Safety Analysis Report
SBGTS	Standby Gas Treatment System
SBLOCA	Small-Break Loss-of-Coolant Accident
SBO	Station Blackout
SE	Safety Evaluation
SER	Safety Evaluation Report
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SFP	Spent Fuel Pool
SLO	Single Loop Operation
SP	Suppression Pool
SPC	Suppression Pool Cooling
SR	Surveillance Requirement

SRLR	Supplemental Reload Licensing Report
SRM	Source Range Monitor
SRP	Standard Review Plan
SRV	Safety Relief Valve
SSCs	Structures, Systems, and Components
STP	Simulated Thermal Power
TAF	Top of Active Fuel
TER	Technical Evaluation Report
TIP	Traversing Incore Probes
TLO	Two Loop Operation
Tmin	Minimum Temperature for Stable Film Boiling
TS	Technical Specification
TTNBP	Turbine Trip Without Bypass
UCP	Upper Containment Pool
UFSAR	Updated Final Safety Analysis Report
V&V	Verification and Validation
W_T	Percent Core Flow
WW	Wetwell

proprietary version of the SE is provided in Enclosure 3. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Alan B. Wang, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures:
1. Amendment No. 205 to NPF-29
2. Safety Evaluation (non-proprietary version)
3. Safety Evaluation (proprietary version)

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