May 17, 2007

Mr. Michael Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT RE: IMPLEMENTATION OF THE AVERAGE POWER RANGE MONITOR, ROD BLOCK MONITOR TECHNICAL SPECIFICATION IMPROVEMENTS WITH THE MAXIMUM EXTENDED OPERATING DOMAIN ANALYSIS (TAC NO. MC9681)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 287 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 26, 2006, as supplemented by letter dated December 21, 2006.

The amendment will allow additional startup and operating flexibility and an expanded operating domain resulting from the proposed implementation of the Average Power Range Monitor, Rod Block Monitor Technical Specification improvement program concurrently with the proposed implementation of the Maximum Extended Operating Domain Analysis, which is the combination of the power/flow operating map expansion with Maximum Extended Load Line Limit Analysis and increased core flow.

Enclosure 2 is the non-proprietary Safety Evaluation. Enclosure 3 is the proprietary Safety Evaluation. The proprietary information is identified by a double underline inside double square brackets. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

/RA/

John P. Boska, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosures:
1. Amendment No. 287 to DPR-59
2. Safety Evaluation (non-proprietary)
3. Safety Evaluation (proprietary)

cc w/encls 1 and 2: See next page

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Accession: ML070430065; Enclosure 3: ML070860247

OFFICE	LPL1-1/PM	LPL1-1/LA	EICB/BC
NAME	JBoska	SLittle	AHowe as signed on
DATE	5/07/07	2/28/07	01/ 31 /07

OFFICE	SCVB/BC	SBWB/BC	ITSB/BC	OGC	LPL1-1/BC
NAME	RDennig as signed on	GCranston as signed on	TKobetz	Myoung	MKowal
DATE	01/ 25 /07	01/ 30 /07	3/30/07	5/14/07	5/16/07

DATED: May 17, 2007

AMENDMENT NO. 287 TO FACILITY OPERATING LICENSE NO. DPR-59 FITZPATRICK

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RidsNrrDirsItsb
RidsNrrDssSbwb
RidsNrrDssScvb
RidsNrrDeEicb
RidsAcrsAcnwMailCenter
RidsNrrPMJBoska
RidsNrrLASLittle
ECobey, RI

cc: Plant Mailing list

FitzPatrick Nuclear Power Plant

cc:

Mr. Gary J. Taylor
Chief Executive Officer
Entergy Operations, Inc.
1340 Echelon Parkway
Jackson, MS 39213

Mr. John T. Herron
Sr. VP and Chief Operating Officer
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Peter T. Dietrich
Site Vice President
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

Mr. Kevin J. Mulligan
General Manager, Plant Operations
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

Mr. Oscar Limpias
Vice President Engineering
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Christopher Schwarz
Vice President, Operations Support
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. John F. McCann
Director, Licensing
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Ms. Charlene D. Faison
Manager, Licensing
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Michael J. Colomb
Director of Oversight
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. David Wallace
Director, Nuclear Safety Assurance
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

Mr. James Costedio
Manager, Regulatory Compliance
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

Assistant General Counsel
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Resident Inspector's Office
James A. FitzPatrick Nuclear Power Plant
U. S. Nuclear Regulatory Commission
P.O. Box 136
Lycoming, NY 13093

cc:

Mr. Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Mr. Peter R. Smith, President
New York State Energy, Research,
  and Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Mr. Paul Eddy
New York State Dept. of Public Service
3 Empire State Plaza
Albany, NY 12223-1350

Oswego County Administrator
Mr. Steven Lyman
46 East Bridge Street
Oswego, NY 13126

Supervisor
Town of Scriba
Route 8, Box 382
Oswego, NY 13126

Mr. James H. Sniezek
BWR SRC Consultant
5486 Nithsdale Drive
Salisbury, MD 21801-2490

Mr. Michael D. Lyster
BWR SRC Consultant
5931 Barclay Lane
Naples, FL 34110-7306

Mr. Garrett D. Edwards
814 Waverly Road
Kennett Square, PA 19348

ENTERGY NUCLEAR FITZPATRICK, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 287
License No. DPR-59

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

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated January 26, 2006, as supplemented on December 21, 2006, 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. DPR-59 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 287, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Mark G. Kowal, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and
Technical Specifications

Date of Issuance: May 17, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 287

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Replace the following page of the License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
3	3

Replace the following pages of 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.

Remove Pages	Insert Pages
Table of Contents i	Table of Contents i
1.1-4	1.1-4
3.2.4-1	- - -
3.2.4-2	- - -
3.3.1.1-3	3.3.1.1-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 287 TO FACILITY OPERATING LICENSE NO. DPR-59

ENTERGY NUCLEAR OPERATIONS, INC.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated January 26, 2006 (Reference 1) (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML060390370), as supplemented by letter dated December 21, 2006 (Reference 3) (ADAMS Accession No. ML063610071), Entergy Nuclear Operations, Inc. (ENO, the licensee) submitted a request for changes to the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Technical Specifications (TSs).

The supplemental letter dated December 21, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination.

The proposed changes would allow additional startup and operating flexibility and an expanded operating domain resulting from the proposed implementation of the Average Power Range Monitor (APRM), Rod Block Monitor (RBM) TS improvement program (ARTS) concurrently with the proposed implementation of the Maximum Extended Operating Domain Analysis (MEOD), which is the combination of the power/flow operating map expansion with Maximum Extended Load Line Limit Analysis (MELLLA) and increased core flow (ICF).

To support this proposed change, the licensee's submittal provided a JAFNPP plant-specific ARTS/MEOD safety analysis report (A/MSAR), NEDC-33087P, Rev. 1 (Reference 2), prepared by the Nuclear Steam Supply System (NSSS) vendor, General Electric Nuclear Energy (GENE). The fuel dependent portions of the safety analyses are based on a representative core with GE12 and GE14 fuel (Cycle 16 core design). Subsequent cycle-specific analyses will be performed by ENO during reload licensing activities. The non-fuel dependent evaluations are based on the JAFNPP plant configuration. All analyses were performed by GENE and used NRC approved and industry-accepted computer codes and calculational techniques.

The JAFNPP is a boiling-water reactor (BWR), 4-series reactor, and the current licensed thermal power (CLTP) is 2536 megawatts-thermal (MWt). The operational flexibility of a BWR during power ascension from the low-power, low-flow core condition to the rated high-power, high-flow core condition is restricted by several factors. Also, once rated power is achieved, periodic adjustments to core flow and control rod positions must be made to compensate for the reactivity changes due to xenon buildup and decay, with fuel and burnable poison burnup. Factors currently restricting plant flexibility at JAFNPP in efficiently achieving and maintaining rated power include:

1. The currently licensed allowable power/flow operating map; and,

2. The current APRM flow-biased flux scram and flow-biased rod block setdown requirements.

The licensee has proposed TS changes to address the above restrictions which are similar to the changes requested and approved at other BWR plants, as discussed in the NRC staff evaluation.

2.0 REGULATORY EVALUATION

The regulatory requirements that the NRC staff considered in its review of the licensing application included those contained in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, “Technical specifications,” which provides the regulatory requirements for the content required in the TSs. 10 CFR 50.36(c)(3) requires that the TSs include surveillance requirements (SR) to assure that facility operation will be within safety limits, and the limiting conditions for operation (LCOs) will be met.

Paragraph (c)(1)(ii)(A) of 10 CFR 50.36, “Technical specifications,” states, in part, that "[w]here a limiting safety system setting [LSSS] is specified for a variable on which a safety limit [SL] has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded." The analytical limit (AL) is the limit on the process variable at which the instrument loop protective action occurs as assumed in the plant's safety analysis. Protective action at the AL ensures that the SL is not exceeded. The AL, however, does not account for uncertainties associated with the instrument loop. The instrument loop uncertainty is accounted for during calculation of an instrument loop's trip setpoint (TSP). Accordingly, limits for instrument channels that initiate protective functions must be included in the TSs. Setpoints found to exceed TS limits are considered a malfunction of an automatic safety system. Such an occurrence could challenge the integrity of the reactor core, reactor coolant pressure boundary, containment, and associated safety systems.

10 CFR 50.62 provides requirements for reduction of risk from anticipated transient without scram (ATWS) events for light-water-cooled nuclear power plants. Requirements for boiling-water reactors include an ATWS Recirculation Pump Trip, an Alternate Rod Injection (ARI) system, and an adequate Standby Liquid Control System (SLCS) injection rate. The GENE safety analysis portion of the licensee’s submittal included a discussion of the plant-specific ATWS analysis performed by GENE to demonstrate compliance with 10 CFR 50.62 after ARTS/MEOD implementation.

10 CFR 50.46 provides acceptance criteria for emergency core cooling systems (ECCS) for light-water power reactors. Appendix K to Part 50 of 10 CFR provides required and acceptable features of ECCS evaluation models. Although the proposed amendment to the JAFNPP

license for implementation of ARTS/MEOD does not explicitly involve changes to the ECCS, the requirements of 10 CFR 50.46(b) are pertinent to the evaluation of acceptability of the proposed amendment. The requirements of 10 CFR 50.46 (b) for maximum fuel element cladding temperature, maximum cladding oxidation, and maximum hydrogen generation during a design-basis loss-of-coolant accident (LOCA) must still be satisfied.

The following explains the use of general design criteria for JAFNPP. The construction permit for JAFNPP was issued by the Atomic Energy Commission (AEC) on May 20, 1970, and the operating license was issued on October 17, 1974. The plant design criteria for the construction phase is listed in the Updated Final Safety Analysis Report (UFSAR) Chapter 1.5, "Principal Design Criteria." The AEC published the final rule that added Appendix A to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "General Design Criteria for Nuclear Power Plants," in the Federal Register (36 FR 3255) on February 20, 1971, with the rule effective on May 21, 1971. In accordance with an NRC staff requirements memorandum from S. J. Chilk to J. M. Taylor, "SECY-92-223 - Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992 (ADAMS Accession No. ML003763736), the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971, which includes JAFNPP. However, the JAFNPP UFSAR, Chapter 16.6, "Conformance to AEC Design Criteria," evaluates JAFNPP against the 10 CFR 50 Appendix A GDC. Also, the initial AEC safety evaluation of JAFNPP, dated November 20, 1972, Chapter 14.0, stated "Based on our evaluation of the design and design criteria for the James A. FitzPatrick Nuclear Power Plant, we conclude that there is reasonable assurance that the intent of the General Design Criteria for Nuclear Power Plants, published in the Federal Register on May 21, 1971 as Appendix A to 10 CFR part 50, will be met." Therefore, the NRC staff reviews amendments to the JAFNPP license using the 10 CFR 50 Appendix A GDC unless there are specific criteria identified in the UFSAR.

General Design Criterion (GDC) 4 of 10 CFR Part 50, Appendix A, "Environmental and dynamic effects design bases," states that "[s]tructures, 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."

GDC 10, "Reactor design," requires that "[t]he reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits [SAFDLs] are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences [AOOs]."

GDC 12, "Suppression of reactor power oscillations," requires that "[t]he 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 16, "Containment design," requires that "[r]eactor 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 50, "Containment design basis," requires that "[t]he 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." Specific review criteria are contained in Standard Review Plan Section 6.2.1.1.c.

Regulatory Guide (RG) 1.105, “Setpoints for Safety-Related Instrumentation,” endorses Part 1 of ISA-S67.04-1994 and describes a method acceptable to the NRC staff for complying with NRC’s regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits. The RG lists four exceptions and clarifications to the standard in regard to crafting an acceptable setpoint methodology. The two exceptions considered for this license amendment were that the LSSS is being specified as a technical-specification-defined limit in order to satisfy the requirements of 10 CFR 50.36 (Exception # 3) and that the allowable value’s relationship to the setpoint methodology and testing requirements in the TSs must be documented (Exception # 4). In addition, the NRC issued a Regulatory Issue Summary 2006-17 on August 24, 2006, which provided the NRC staff position on the requirements of 10 CFR 50.36, “Technical specifications,” regarding limiting safety system settings during periodic testing and calibration of instrument channels.

The proposed TS changes would revise SRs and the LCO actions and completion times for each applicable operating condition, and are consistent with the requirements of NUREG-1433, "Standard Technical Specifications - General Electric Plants, BWR/4," Revision 3. The NRC has previously approved similar amendments for plants, such as LaSalle County Station, Units 1 and 2 (References 4 and 5), Dresden Nuclear Power Station, Units 2 and 3 (Reference 6), Quad Cities Nuclear Power Station, Units 1 and 2 (Reference 7), Vermont Yankee Nuclear Power Station (Reference 8), and Hope Creek Generating Station (References 9 and 10).

3.0 TECHNICAL EVALUATION

3.1 Systems Analysis

3.1.1 Background

The function of the allowable power/flow operating map is to define the normal operating condition of the reactor core used in determining the operating safety limits. The current approved operating domain for JAFNPP is the Extended Load Line Limit Analysis (ELLLA) map (Reference 11). The proposed TS change reflects operation of JAFNPP in a region which is above the current rated rod line. The power/flow operating map includes the operating domain changes for ARTS/MEOD consistent with NRC approved operating domain improvements for other BWRs. This performance improvement program expands the operating domain along the approximate 116% rod line to 100% CLTP at approximately 80% of rated core flow and increases maximum core flow to 105%. This domain is defined in Figure 1-1 of the GENE A/MSAR (Reference 2). Excluding the ICF portion, this extended operating domain is called the Maximum Extended Load Line Limit (MELLL). The analyses for the specific operating limits associated with the MELLL region, are referred to as MELLLA. The conclusions will be revalidated on a cycle-specific basis as part of the reload licensing scope.

The function of the RBM is to prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high-power level operation. It does this by blocking control rod movement which could result in violating a thermal limit (the safety limit Minimum Critical Power Ratio or the 1% cladding plastic strain limit) in the event of a Rod Withdrawal Error (RWE) event.

The functions of the APRM system include:

1. Generate trip signals to automatically scram the reactor during core-wide neutron flux transients before the actual core-wide neutron flux level exceeds the safety analysis design bases. This prevents exceeding design bases and licensing criteria from single operator errors or equipment malfunctions.

2. Block control rod withdrawal whenever operation occurs in excess of set limits in the operating map and before core power approaches the scram level.

3. Provide an indication of the core average power level of the reactor in the power range.

The flow-biased rod block setdown and APRM flow-biased flux scram trip and alarm functions are provided to achieve these requirements.

The proposed implementation of the ARTS/MEOD improvement program will increase the plant operating efficiency by updating the thermal limits requirements to be consistent with current General Electric (GE) methodology and by introducing improvements in plant instrumentation accuracy, such as the new APRM digital flow control trip reference (FCTR) cards.

The ARTS improvement program includes changes to the current APRM system, which requires the TS changes described in Section 3.10 of this safety evaluation. The current JAFNPP TSs require that the plant operate such that the core Maximum Fraction of Limiting Power Density (MFLPD), which is equivalent to the ratio of Maximum Total Peaking Factor (MTPF) to the Design Total Peaking Factor (DTPF), does not exceed the Fraction of Rated Power (FRP). This requirement limits the maximum local power at lower core power and flows to a fraction of that allowed at rated power and flow. If the MTPF exceeds the DTPF, the flow-referenced APRM trips must be lowered (setdown) to limit the maximum power that the plant can achieve. The basis for this “APRM trip setdown” requirement originated from now obsolete Hench-Levy Minimum Critical Heat Flux Ratio thermal limit criterion (Reference 12) and provides conservative restrictions with respect to current fuel thermal limits. A subsequent update to the thermal limits requirements, which decreases the dependence on the local thermal hydraulic conditions, including the core peaking factors, was developed by GE. The resulting General Electric Thermal Analysis Basis (GETAB) critical power ratio (CPR) correlation model (Reference 13), which relies on bundle boiling length and exit quality, was reviewed and approved by the NRC staff.

The objective of the APRM improvements is to justify removal of the APRM trip setdown and DTPF requirement. Since the elimination of the APRM gain and setpoint requirement can potentially affect the fuel thermal-mechanical integrity and the emergency core cooling system (ECCS) LOCA performance, the NRC staff reviewed the acceptability of these changes. The following criteria are used to assure satisfaction of the applicable licensing requirements to demonstrate acceptability of the APRM trip setdown requirement:

1. The safety limit minimum critical power ratio (SLMCPR) shall not be violated as a result of any AOOs,

2. All fuel thermal-mechanical design bases shall remain within the licensing limits described in the GE generic fuel licensing report (GESTAR-II), and,

3. The peak cladding temperature and the maximum cladding oxidation fraction following a LOCA shall remain within the limits defined in 10 CFR 50.46.

The ARTS-specific changes are summarized here:

1. The requirement for setdown of the APRM scrams and rod blocks is deleted,

2. New power-dependent minimum critical power ratio (MCPR) adjustment factors, MCPR(p), are added,

3. New flow-dependent MCPR adjustment factors, MCPR(f), are added,

4. New power-dependent maximum average planar linear heat generation rate (MAPLHGR) and linear heat generation rate (LHGR) adjustment factors, MAPFAC(p) and LHGRFAC(p) respectively, are added,

5. New flow-dependent MAPLHGR and LHGR adjustment factors, MAPFAC(f) and LHGRFAC(f) respectively, are added, and

6. The affected TS SRs, LCOs, and the associated Bases are modified or deleted, as required.

The NRC staff reviewed the safety analyses and systems response evaluations performed by GENE to justify JAFNPP operation in the expanded MEOD region, as discussed in Reference 2. The plant-specific, fuel independent evaluations, such as containment response, were performed based on the current hardware design and applicable plant geometry for JAFNPP. The fuel dependent analyses, such as the limiting AOOs, the MCPR calculations, and the reactor vessel overpressure protection analysis, were performed. A representative core of GE12 and GE14 fuel types was used for the basis of the analysis. These analyses are to be performed each operating cycle as part of the standard reload design process, outlined in the current version of GESTAR-II (Reference 14).

3.1.2 Method of Analysis

The ARTS transient analyses were performed at the CLTP plant conditions for the current Cycle 16 core configuration, using the GENE standard reload licensing methodology described in the GESTAR-II documentation (Reference 14). The JAFNPP plant-specific evaluations were performed to establish plant-unique, flow-dependent MCPR, LHGR, and MAPLHGR limits. Added conservatisms were included which are expected to allow future reloads of GE14 fuel design.

The NRC staff finds the licensee's method of analysis for the JAFNPP MEOD operation acceptable, as the analysis methodology used (Reference 14) has been previously approved by the NRC, with the numerous NRC safety evaluations listed in the approved version of Reference 14 (ADAMS accession number ML011230173), and the results meet the intent of the GDC 10 criteria.

3.1.3 Fuel Thermal Limits

Extensive transient analyses at a variety of power and flow conditions were performed during original development of the ARTS improvement program. These evaluations are applicable for operation in the ARTS/MELLLA region. The analyses were utilized to study the trend of transient severity without the APRM trip setdown. A database was established by analyzing limiting transients over a range of power and flow conditions. The database included evaluations representative of a variety of plant configurations and parameters such that the conclusions drawn from the studies would be applicable to all BWRs. The database was utilized to develop a method of specifying plant operating limits (MCPR, and LHGR or MAPLHGR), which ensures that margins to fuel safety limits are equal to or larger than those applied currently.

The NRC staff reviewed the effects of operation in the MEOD region at the CLTP on the thermal limits and the thermal limits management with the ARTS power and flow dependent limits, which are covered in the A/MSAR.

The potentially limiting AOOs and accident analyses were evaluated to support JAFNPP operation with the ARTS off-rated limits, as well as operation in the MEOD region for Cycle 16 fuel and core configuration at 100% of CLTP. The power/flow state points chosen for the review of the AOOs listed in Table 3-1 of Reference 2 bound the current licensed operating domain and the ARTS/MEOD region. The minimum core flow at 100% of rated thermal power (RTP) used in the analysis presented in this section is 81% of rated core flow (RCF). Figure 1-1 of Reference 2 shows this point as 79.8% of RCF. In a request for additional information, the NRC staff questioned why the licensee did not use the exact point at 79.8% of RCF corresponding to Figure 1-1 in the analysis. The licensee responded that there is a minimal effect on the results of the fuel thermal limits analysis due to this difference in minimum rated core flow. Subsequent work for cycles 17 and 18 (the current cycle) has been performed at the actual minimum flow of 79.8%. The conclusion from the transient analyses reported in NEDC-33087P is that JAFNPP can operate within standard off-rated ARTS limit curves. That conclusion is not affected by the small difference in assumed minimum core flow. The NRC staff concluded that the licensee's arguments are reasonable, and that the licensee's conclusion addressed the staff's concerns raised in the request for additional information.

The core-wide AOOs included in the JAFNPP Cycle 16 reload licensing analyses and the JAFNPP Updated Final Safety Analysis Report (UFSAR) (Reference 15) were re-examined by the licensee for operation in the ARTS/MEOD region (including off-rated power and flow conditions). The following events were considered potentially limiting in the ARTS/MEOD region and were reviewed as part of the ARTS program development:

1. Generator Load Rejection with No Bypass (LRNBP) event;

2. Turbine Trip with No Bypass (TTNBP) event;

3. Feedwater Controller Failure (FWCF) maximum demand event;

4. Loss of Feedwater Heating (LFWH) event;

5. Fuel Loading Error (FLE) event;

6. Idle Recirculation Loop Start-up (IRLS) event; and

7. Fast Recirculation Flow Increase (FRFI) event.

The initial ARTS/MELLLA assessment of these events for all BWR type plants concluded that for plant-specific applications, only the TTNBP, LRNBP, and FWCF events need to be evaluated at both rated and off-rated power and flow conditions. The LFWH evaluation at 81% flow for JAFNPP Cycle 16, showed that there is a large margin in the Operating Limit Minimum Critical Power Ratio (OLMCPR) for the LFWH event compared to the LRNBP event. Considering that the LFWH event tends to become less limiting as the power decreases (less feedwater to be affected by loss of heating), the LFWH event was not considered in the determination of the off-rated limits. The FLE event is most limiting at maximum power; thus, this event was also not considered in the determination of the off-rated limits. The other two events (IRLS and FRFI) are generally most limiting at off-rated conditions. Even when originated from their most limiting off-rated condition, the IRLS and FRFI are less limiting than the fast pressurization events (TTNBP, LRNBP, or FWCF) at rated power conditions. Thus, the IRLS and FRFI events were not considered in the determination of the off-rated limits.

The generic evaluations determined that the power-dependent severity trends must be examined in two power ranges. The first power range is between rated power and the power level, P(bypass), where reactor scram on turbine stop valve closure or turbine control valve fast closure is bypassed. P(bypass) for JAFNPP is 29% of CLTP. The second power range is between P(bypass) and 25% of CLTP. No thermal monitoring is required below 25% of CLTP per JAFNPP TSs.

Generic power-dependent MCPR, and LHGR or MAPLHGR limits (in terms of multipliers on the plant's rated operating limits) were developed for use in the power range between rated power and P(bypass). Below P(bypass), the OLMCPR is specified while the MAPLHGR (or LHGR) retains the use of a multiplier on the rated limits. Generic flow-dependent MCPR and MAPLHGR (or LHGR) limits were also developed from the ARTS database.

JAFNPP specific analyses were performed to confirm the applicability of the generic power dependent limit multipliers [K(p), LHGR(p), and MAPLHGR(p)] above P(bypass). JAFNPP specific evaluations were also performed between P(bypass) and 25% power to establish

JAFNPP-unique MCPR, LHGR, and MAPLHGR limits. JAFNPP specific evaluations were also performed to establish the flow-dependent MCPR, LHGR, and MAPLHGR limits.

The AOO analyses were performed for operating Cycle 16 fuel and core configuration at 100% of CLTP with the MEOD operating power/flow statepoints, generating operating limits for the Cycle 16 core. For AOOs, cycle specific analyses are performed for the limiting transients. These transient analyses use the cycle specific nuclear and thermal-hydraulic characteristics of the core to establish the rated and off-rated power operating limits for the fuel types that comprise the core. These transient analyses also consider the ARTS/MEOD operating domain for establishing initial conditions for the transient initiation.

The partial ARTS improvement implementation at JAFNPP will not require the original ARTS hardware change to the RBM that provided protection for an off-rated RWE event. Therefore, evaluation of the JAFNPP RWE event is performed without taking credit for the mitigating effect of the flow-biased RBM setpoints, and the resulting off-rated OLMCPR values are for the unblocked RWE event. This is consistent with the implementation of the partial ARTS program at the other BWRs, such as Hope Creek Generating Station. The cycle specific (Cycle 16) analysis was performed for JAFNPP plant-specific power-dependent RWE OLMCPR value. If, for future reload cycle operating conditions, the unblocked RWE event OLMCPR values are too restrictive, the RBM setpoint adequacy would be readdressed.

In A/MSAR (Reference 2), Single Loop Operation (SLO) was discussed and the NRC staff raised questions regarding SLO in the ARTS/MEOD operating domain. The licensee clarified that there is no change to SLO associated with this TS amendment application. MEOD boundaries apply to two-loop operation and do not affect SLO.

The cycle-specific reload fuel analyses will determine the limits for rated and applicable off-rated conditions, and application of the methodology is demonstrated by the analyses performed for the Cycle 16 core. This approach is acceptable to the NRC staff, as JAFNPP TS 5.6.5, "Core Operating Limits Report," requires that core operating limits be established prior to each reload cycle using analytical methods previously reviewed and approved by the NRC.

3.1.4 Reactor Recirculation System

The licensee has requested to increase maximum core flow to 105%. The reactor recirculation (RR) system was evaluated for ICF conditions. The evaluation included the suction and discharge pressure and temperature, pump speed, drive flow, and head requirements, and the drive motor current and power requirements. Evaluations indicate that the capability of the RR system to support operation at 105% of RCF may be marginal during some of the fuel cycle, due to rotating equipment limitations that are economic and do not affect plant safety. Currently the licensee uses stops on the RR motor generator subsystem to prevent exceeding 100% core flow. These stops will be reset to allow 105% flow upon implementation of this amendment, although, as noted above, the RR system may not always be able to achieve 105% flow.

The feedwater flowing into the reactor vessel annulus during operation provides subcooling for the fluid passing to the recirculation pumps, thus providing the additional net positive suction head (NPSH) available beyond that provided by the pump location below the reactor vessel water level. If feedwater flow is below about 20 percent, the recirculation pump speed is automatically limited. The recirculation pump net positive suction head (NPSH) requirements increase in the ICF region. The licensee did not recommend increasing the setpoint of this automatic cavitation protection interlock as that would adversely affect plant maneuverability as represented by the power/flow map. There were no physical changes proposed to the RR system other than resetting the stops. There is no effect on plant safety and, therefore, operation of the RR system as described is acceptable to the NRC staff.

3.1.5 Reactor Coolant Pressure Boundary

The licensee evaluated the reactor coolant pressure boundary (RCPB) piping system including associated branch piping inside containment to determine its structural integrity under the MEOD operating conditions. MEOD conditions primarily affect the pressures, temperatures, and flows for the following RCPB piping systems: RR system, reactor pressure vessel (RPV) Bottom Head Drain Line system, and their associated branch piping in containment. The evaluation also included piping supports and interfacing piping system components. For these affected systems, the MEOD conditions were compared to the existing piping analysis basis and/or current limiting values and the current analysis is based on higher or more limiting values than the actual MEOD conditions entail. Therefore, the current analysis is adequate and acceptable to the NRC staff.

3.1.6 Vessel Overpressure

A cycle-specific overpressure analysis of Cycle 16 was performed at 102% of CLTP with 81% flow and 105% flow. The licensee showed that the peak steam dome pressure, peak steam line pressure, and peak vessel pressure are below ASME Code limits of 1375 psig. Therefore, the analysis results are acceptable to the NRC staff.

3.1.7 Thermal-Hydraulic Stability

Stability criteria are established in GESTAR-II to demonstrate compliance with the GDC 12 requirements in order to assure that specified acceptable fuel design limits (i.e., SLMCPR) are not exceeded. The analysis and methods used to demonstrate compliance with the stability acceptance criteria are documented in NEDO-31960A (Reference 16).

The licensee has implemented Option 1D (Reference 16) as the long-term solution, which is only applicable to plants that can demonstrate that core wide mode instability is the predominant mode and regional mode instability is not expected.

Option 1D has prevention elements, such as exclusion and buffer regions, and a detect and suppress element. Application of the Option 1D solution consists of calculating an administratively controlled exclusion region (per Reference 2) and demonstrating that the existing flow-biased APRM flux trip line provides adequate SLMCPR protection. Both the Option 1D exclusion region and flow-biased APRM flux trip protection are fuel cycle dependent.

To determine the exclusion and buffer regions for the Cycle 16 stability analysis, the licensee applied the NRC-approved ODYSY methodology (Reference 17). This analysis demonstrated the core-wide instability mode is the predominant one and that the exclusion region is affected by operating conditions. This confirms that the licensee should be using Option 1D as its stability solution. The actual regions will be calculated using the ODYSY code for future operating cycles. To calculate the detect and suppress element, the licensee determined the 95% probability / 95% confidence level (95/95) statistically-based hot bundle oscillation magnitude for anticipated core-wide mode reactor instability and determined the stability-based OLMCPR which provides 95/95 SLMCPR protection. This calculation required the use of the Delta CPR over Initial MCPR Versus the Oscillation Magnitude (DIVOM) curve (Reference 18). A non-conservative deficiency has been identified for Option 1D plants with a high power-to-flow ratio. Licensees using the ODSYS methodology must perform a calculation in order to determine if the DIVOM curve is applicable to the licensee's requested power-to-flow ratio. The licensee calculated its ratio to be 56.8 MWt/Mlbm/hr, which is below the required 66 Mwt/Mlbm/hr. Since the licensee's ratio did not exceed the required ratio, the DIVOM curve is applicable for this cycle, but this value should be calculated for each core reload in order to determine its applicability to each cycle.

With the new operating conditions with application of ARTS/MEOD, a new APRM flow-biased flux scram line was determined with the following additional conservatisms in the evaluation:

1. [[____ __________]];

2. The SLMCPR is assumed to be 1.12, and the OLMCPR is assumed to be 1.35; and

3. [[____ __________________]].

The detect and suppress evaluation then showed that adequate SLMCPR protection is provided by the new APRM flow-biased flux scram line. This stability-based scram line provides more than 5% rated core power margin at the most limiting state point, which is the intersection of the Natural Circulation Line and the High Flow Control Line, and this will avoid a spurious scram due to a single pump trip. The licensee is also implementing the conservative APRM flow-biased scram line for all power levels. From recirculation drive flows greater than 47% and up to 68.7%, the licensee uses non-stability based APRM flow-biased flux scram line. The maximum AV clamps at 117% power for recirculation drive flows greater than 68.7%. This approach is conservative and acceptable to the NRC staff, as it meets the criteria of GDC 12 by protecting against reactor power oscillations.

Based on the analyses provided by the licensee and the fact that approved methodologies were used, the NRC staff concludes that the thermal hydraulic stability characteristic of the JAFNPP with the proposed ARTS/MELLLA implementation at the CLTP conditions is acceptable.

3.1.8 LOCA Analysis

The ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The MAPLHGR operating limit is based on the most limiting LOCA and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46.

The JAFNPP SAFER/GESTR-LOCA analysis was performed with GE14 fuel to determine the effects on the LOCA analysis of operation in the MELLLA domain. The Reference 19 analysis for GE12 fuel was performed for JAFNPP operation in the ELLLA domain. The JAFNPP SAFER/GESTR-LOCA methodology was used consistent with Reference 19. Since no evaluation was performed for GE12 fuel in the MELLLA domain, a conservative analysis was performed as described in the next paragraph. Also, the effects of ICF operation on the ECCS-LOCA analysis results are negligible because there is no impact on fuel dryout time and there is a higher core flow, which delays the onset of boiling transition. The evaluation was performed with the limiting GE14 fuel to determine the effects in the MELLLA domain. The limiting design-basis accident (DBA) was evaluated to show the effect on the peak clad temperature (PCT) based on limiting MELLLA conditions. These results show that operation in the MELLLA region affects the nominal PCT by +3 °F and the Appendix K PCT by +93 °F.

The Upper Bound PCT is most directly related to nominal PCT and therefore, +3 °F does not significantly affect this PCT. With the +93 °F change in Appendix K PCT, the JAFNPP licensing basis PCT for GE14 fuel is 1700 °F, which still leaves 500 °F of margin to the 2200 °F licensing limit. This +93 °F PCT was also applied conservatively to the GE12 results. The current JAFNPP licensing basis PCT for GE12 fuel is 1370 °F with a 170 °F conservative adder for 10 CFR 50.46 reported errors applicable to JAFNPP ECCS-LOCA analysis. In addition, adding the +93 °F due to MELLLA conditions still leaves greater than 500 °F margins to the 2200 °F licensing limit for GE12 fuel.

The PCT for a large recirculation line break is affected by MELLLA because the core flow is reduced in the MELLLA range, which leads to earlier boiling transition at lower elevation in the fuel bundle. For small breaks, however, the fuel remains in nucleate boiling until uncovery; therefore, MELLLA does not have an adverse effect on the small-break LOCA response. MELLLA has a negligible effect on compliance with the other acceptance criteria of 10 CFR 50.46. The NRC staff has, therefore, determined that no additional operating restrictions would be required for ARTS/MEOD operation at the CLTP, since the determination of the sensitivity of the ECCS-LOCA evaluations to operation in the MELLLA domain shows compliance with the acceptance criteria in 10 CFR 50.46.

3.1.9 Anticipated Transient without Scram (ATWS)

The basis for the current ATWS requirements is 10 CFR 50.62. This regulation includes requirements for an ATWS Recirculation Pump Trip (RPT), an Alternate Rod Injection (ARI) system, and an adequate Standby Liquid Control System (SLCS) injection rate.

The NRC staff reviewed the JAFNPP specific analysis that was performed using the approved licensing methodology (Reference 20) to demonstrate compliance with 10 CFR 50.62 ATWS requirements. The analysis assumed the CLTP with the minimum MELLLA core flow (80% of RCF), which is the limiting operating condition. The limiting ATWS events, main steam isolation valve (MSIV) closure and pressure regulator failure open (PRFO), were re-evaluated with ARI assumed to fail, requiring SLCS injection. The adequacy of the margin to the SLCS relief valve lifting, as described in NRC Information Notice 2001-13, “Inadequate Standby Liquid Control

System Relief Valve Margin," was included in this assessment. Two safety/relief valves (SRVs) were assumed to be out of service (OOS), which were specified as the valves with the lowest setpoints.

The limiting ATWS event for JAFNPP is the PRFO at beginning of cycle (BOC). The peak vessel bottom pressure response for this limiting event is below the ATWS vessel overpressure protection criterion of 1500 psig. Therefore, the vessel overprotection criterion for ATWS is satisfied. Also, suppression pool temperature and containment pressure were satisfied for both BOC and end of cycle (EOC). These results are shown in Tables 11-3 and 11-4 in Reference 2, respectively.

The NRC staff concludes, based on its review of the above analyses, that JAFNPP meets the ATWS mitigating features stipulated in 10 CFR 50.62 and that the results of the ATWS analyses for MELLLA operation at the CLTP would meet the ATWS acceptance criteria for JAFNPP Cycle 16.

3.2 Instrumentation Analysis

3.2.1 Flow-Biased APRM Scram and Rod Block Design Bases

The licensee has stated that for the cycle 16 licensed power/flow map, the flow-biased APRM scram allowable value (AV) was defined as 0.58Wd+66%, clamped at 117% , where Wd is the recirculation drive flow in percent of rated, and where 100% drive flow is that required to achieve 100% core power and flow. The flow-biased APRM rod block line AV was set at 0.58Wd+54%, with no clamp. The RBM flow-biased AV was set at 0.66Wd+42%, clamp at 110%. To accommodate the proposed power/flow map expansion to include the MEOD region, the upper boundary of the licensed operating domain is extended. To accommodate this and to restore the pre-existing margin between the MEOD boundary line and the flow-biased APRM rod block line and to ensure compliance with long-term thermal-hydraulic stability solution, the following AVs have been redefined for two loop operation (TLO) and for single loop operation (SLO):

(a) APRM flow-biased scram AVs for TLO are:

0.38Wd+61%	for $0<Wd\leq24.7\%$
1.15Wd+42%	for $24.7<Wd\leq47.0\%$
0.63Wd+73.7%	for $47.0<Wd\leq68.7\%$

Clamp at 117% of current licensed thermal power (CLTP) for $Wd>68.7\%$

(b) APRM flow-biased rod block AVs for TLO are:

0.38Wd+49%	for $0<Wd\leq24.7\%$
1.15Wd+30%	for $24.7<Wd\leq47.0\%$
0.63Wd+61.7%	for $47.0<Wd\leq78.3\%$

Clamp at 111.0% of CLTP for $Wd>78.3\%$

(c) APRM flow-biased scram AVs for SLO are:

0.38Wd+57.9%	for $0<Wd\leq32.7\%$

1.15Wd+32.8% for $32.7 < Wd \leq 50.1\%$
0.58Wd+61.3% for $50.1 < Wd \leq 95.9\%$
Clamp at 117% of CLTP for $Wd > 95.9\%$

(d) APRM flow-biased rod block AVs for SLO are:

0.38Wd+45.9% for $0 < Wd \leq 32.7\%$
1.15Wd+20.8% for $32.7 < Wd \leq 50.1\%$
0.58Wd+49.3% for $50.1 < Wd \leq 106.3\%$
Clamp at 111.0% of CLTP for $Wd > 106.3\%$

The AV for the RBM flow-biased rod block is being revised to 0.66Wd+K, with clamp at 125%, where K is a constant determined by setpoint calculations. The licensee has justified the relaxation for the RBM setpoint to allow full MEOD operation based on the fact that they do not take any credit for RBM in the rod withdrawal error for the reload analysis and therefore, the RBM clamp and normal rod block can be set at the highest values that can be calibrated in the hardware.

The licensee has performed an analysis to demonstrate that (a) the safety limit minimum critical power ratio (MCPR) shall not be violated as a result of any anticipated operational occurrences; (b) all fuel thermal-mechanical design bases shall remain within the licensing limits described in the GE generic licensing report GESTAR-II; and (c) peak cladding temperature and maximum cladding oxidation fraction following a LOCA shall remain within the limits defined in 10 CFR 50.46. These analyses have been reviewed and accepted by the NRC staff. The licensee has proposed to use the same AVs for the APRM flow-biased scram as are currently in use at the plant. These AVs are set by the licensee's Core Operating Limits Report (COLR). For example, see revision 22 of the COLR, ADAMS accession number ML070030006. The licensee controls the COLR under TS 5.6.5, using analytical methods previously reviewed and approved by the NRC. Therefore, the NRC staff finds the proposed changes acceptable.

3.2.2 Acceptability of Digital Flow Control Trip Reference (FCTR) Card

In its submittal dated January 26, 2006, the licensee stated that the analog FCTR cards are being replaced with digital FCTR cards. The current JAFNPP APRM flow-biased setpoints are implemented by an analog FCTR card installed in each of the APRM channels. These cards are able to accommodate a single flow-biased scram equation. Therefore, to implement the multiple APRM flow-biased equations the licensee has replaced these cards with digital FCTR cards. During the NRC staff's review, the staff requested the licensee to justify the use of digital FCTR cards. The licensee, in its letter dated December 21, 2006, provided the basis for acceptance of these cards. The NRC staff has previously accepted GE Topical Report NEDC-32339P-A, Supplement 2, Revision 1 on the condition that the licensee meets certain design and installation guidance. The licensee's letter dated December 21, 2006, identified that GE document DRF 0000-0027-8895, Qualification Summary - Digital Flow Control Trip Reference (FCTR) Card 148C7596G013 for JAFNPP, Revision 1, September 3, 2004, addresses the applicable design and installation criteria. The NRC staff has reviewed the document and based on the documentation provided by the licensee has determined that the JAFNPP FCTR cards meet the applicable design and installation criteria.

3.2.3 Summary

The NRC staff has reviewed the licensee's proposed changes to setpoints and AVs and the supporting information. Based on its review, the staff concludes that there is reasonable assurance that the plant will operate in accordance with the safety analysis and will continue to meet its licensing basis. Therefore, based on its review, the staff finds that the proposed changes are acceptable.

3.3 Containment Response

As part of its MEOD implementation, the licensee evaluated its effect on the containment pressure and temperature response and on the containment hydrodynamic loads.

The JAFNPP safety design objectives are listed in the UFSAR. Section 5.2.2, which discusses the containment design basis includes the following functional capabilities related to this review:

1. The Primary Containment System has the capability of withstanding the conditions which could result from any of the postulated design-basis accidents for which the Primary Containment System is assumed to be functional, including the largest amount of energy release and mass flow associated with the accident; and has the capability to maintain system integrity indefinitely.

2. The Primary Containment System, which is designed in accordance with seismic Class I design criteria, has the capability to maintain its functional integrity during any of the postulated internal, external, or environmental events.

To demonstrate that the above criteria are satisfied for MEOD implementation, the licensee evaluated the containment short-term pressure and temperature response, long-term pressure and temperature response, and containment hydrodynamic loads assessment for a design-basis LOCA.

In the short-term and long-term containment response analysis, the peak pressure and peak temperature remained below the design values stated in UFSAR Table 5.2-1, "Primary Containment System Principal Design Parameters and Characteristics." This ensures that design basis functional capabilities one and two (above) are met. Containment hydrodynamic loads remained below the JAFNPP MARK I containment unique load definition, thus ensuring that design basis functional capabilities one and two are met.

3.3.1 Short-Term Containment Response

The licensee performed a short-term design-basis LOCA containment performance analysis which covered the blowdown period during which the maximum drywell pressure, drywell temperature and maximum drywell-to-wetwell differential pressure occurred. The analysis was performed for four cases which conservatively covered the full extent of MEOD power/flow boundary. The power and flow values for the four cases analyzed were:

Case 1 - At 102% of CLTP and 100% of Rated Core Flow (RCF).

Case 2 - At 102% of CLTP with 105% of RCF or ICF.

Case 3 - At 102% of CLTP, 79.8% of RCF (on the MELLLA line)

Case 4 - At 62% of CLTP, 36.8% of RCF (minimum recirculation pump speed on the MELLLA line).

The licensee indicated that Cases 1, 2, and 3 were analyzed with Normal Feedwater Temperature (NFWT) and with Final Feedwater Temperature Reduction (FFWTR). Cases with FFWTR assumed a feedwater temperature reduction of 100 °F. Case 4 was analyzed with NFWT only. [[____]].

When evaluating the short-term containment pressure and temperature response for the design-basis LOCA, the licensee used the same assumptions that are standard for MARK I containment design-basis LOCA analysis with the General Electric (GE) M3CPT containment code with LAMB code generated break flows using the 'slip break flow model'. The M3CPT and LAMB code have been previously accepted by NRC for use in LOCA containment analysis as noted in Table 1-1 of Reference 2. The licensee's sensitivity study showed that both peak drywell pressure and temperature for the design-basis LOCA are bounding for case 2 condition with NFWT. The peak drywell pressure and temperature for case 2 with NFWT were 58.37 psia and 290.24 °F, respectively, and peak drywell-to-wetwell differential pressure was 25.82 psid. For case 1 with NWFT, the peak drywell pressure and temperature were 58.30 psia and 290.16 °F, respectively, and peak drywell-to-wetwell differential pressure was 25.81 psid. A comparison of case 1 and case 2 results shows that operation with ICF with NWFT results in an insignificant increase in the containment parameters.

The licensee performed a confirmatory calculation for the limiting condition, i.e., case 2 at NWFT established by the above sensitivity study and designated it as case 5 - defined as follows:

Case 5 - At 102% of CLTP, 105% of RCF (or ICF), analyzed with NWFT, and with nominal initial drywell pressure of 16.50 psia, and nominal initial wetwell pressure of 14.85 psia.

The confirmatory calculation was performed using the GE M3CPT code and LAMB code generated break flow based on a different model that ensures a conservatively high blowdown flow. The peak drywell pressure and temperature in this calculation were 53.15 psia and 282.51 °F, respectively. The results show that the peak values of the containment pressure and temperature are within the design limits of 56 psig (70.7 psia) and 309 °F, respectively, given in UFSAR Table 5.2-1. The NRC staff finds this analysis acceptable as it uses NRC-approved methodologies and the results do not exceed the containment design limits.

3.3.2 Long-Term Containment Response

The licensee has stated that the long-term containment pressure and temperature response is not affected by MEOD operation. Because the MEOD operation does not increase the reactor power level nor the vessel operating pressure, neither the decay heat nor the vessel sensible energy is increased. The peak wetwell pressure and temperature and peak suppression pool temperature occur later in the design-basis LOCA and are established by the long-term release of the decay heat and sensible energy from the reactor vessel to the containment and therefore are not impacted by the MEOD operation. The NRC staff finds the licensee's conclusions acceptable for the long-term containment response.

3.3.3 Containment Hydrodynamic Loads

LOCA Loads

The licensee performed the assessment of design-basis LOCA containment hydrodynamic loads based on the short-term containment pressure and temperature response analysis with MEOD operation. The previous load assessment included the following loads: Vent Thrust, Pool Swell, Condensation Oscillation (CO) and Chugging (CH). These loads have been defined generally for the MARK I containment in Reference 21 and was approved by NRC in Reference 22. The present containment loads evaluation extends it by considering the MEOD and FFTWR.

The licensee evaluated the containment hydrodynamic loads using the short-term containment pressure and temperature response. The licensee's evaluation showed that the effects of MEOD operation on the containment are bounded by current JAFNPP MARK I containment unique load definition and remain within the current design basis. Therefore, the licensee concluded that there is no significant effect on the containment design-basis LOCA hydrodynamic loads. Based on the above, the NRC staff finds that a design-basis LOCA during MEOD operation will result in containment hydrodynamic loads that are within the current design basis and are therefore acceptable.

Safety/Relief Valve (SRV) Actuation Loads

The licensee states that the controlling parameters for SRV loads are (1) SRV discharge lines and containment geometry, (2) initial water leg length in the SRV discharge lines for first SRV actuation, (3) reflood length in the SRV discharge lines for subsequent SRV actuations, (4) SRV flow rates, and (5) time interval between SRV openings. The licensee further states that all of these parameters do not change. Therefore, the loads due to initial or subsequent actuations are not affected by MEOD operation. The NRC staff finds that, as there is no change in the licensed power level and no physical changes to the SRVs, the SRV actuation loads will not be affected by MEOD operation and are therefore acceptable.

3.3.4 Summary

Based on the above evaluations, the NRC staff concludes that MEOD operation does not result in an increase in the peak design-basis LOCA drywell pressure and temperature from the design limits in the JAFNPP UFSAR, nor result in conditions that would produce higher containment hydrodynamic loads during a design-basis LOCA than the JAFNPP MARK I containment unique load definition. Therefore, the NRC staff finds that the containment response to a design-basis LOCA while operating in the MEOD region of the power/flow operating map is acceptable.

3.4 TS Changes for ARTS/MEOD

10 CFR 50.36, “Technical specifications,” provides the regulatory requirements for the content required in the TSs. 10 CFR 50.36 requires that the TSs include SRs to assure that the LCO will be met.

The NRC staff reviewed the proposed changes to the JAFNPP TSs. The changes include deletion of the current setdown requirement and revision of the SR reference to the setdown requirement TS. The proposed TS changes are:

TS Section 3.2.4 : "Average Power Range Monitor (APRM) Gain and Setpoint," which includes requirements for flow biased APRM scram and rod block trip setpoint setdown, and the associated TS Bases, was deleted. The following additional changes were made to reflect the deletion of TS 3.2.4:

1. SR 3.3.1.1.2 of TS 3.3.1.1 was revised to delete words associated with TS 3.2.4.

2. Definition of "Maximum Fraction of Limiting Power Density (MFLPD)" was deleted from TS Section 1.1.

3. The TS Index was revised.

These changes allow the implementation of the thermal limits portion of the GE ARTS improvement protection program and the MEOD expanded operating domain. The safety analyses presented examined the same areas as previous ARTS and MEOD or MELLLA submittals, which have been reviewed by the NRC staff. The methods used have been previously approved as noted in Table 1-1 of Reference 2, and the results of the analyses fall within accepted limits. The NRC staff concludes that the results submitted by the licensee justify the proposed TS changes to the JAFNPP for operation at the CLTP, based on the analyses reviewed.

The licensee also provided the associated TS Bases that reflect the proposed TS changes as an attachment to its application. The TS Bases changes are consistent with the licensee’s proposed plant-specific TS changes, and the NRC staff has no objections to the Bases changes presented in the licensee's application.

3.5 Summary

The NRC staff has reviewed the licensee’s application along with the supporting documentation, including responses to the NRC staff’s request for additional information. The review of TS changes in this safety evaluation was performed to evaluate the changes that would be required to support the ARTS/MEOD implementation at JAFNPP. This review covered the ARTS/MEOD application for the CLTP.

Based on its review, the NRC staff concludes that the proposed TS changes are acceptable because the safety analyses supporting actual operation in the ARTS/MEOD regimes at the CLTP has been reviewed and the NRC staff concludes that the proposed operation will not endanger the public health and safety.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 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 and changes surveillance requirements. 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 no public comment on such finding (71 FR 13171). Accordingly, the amendment meets 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.

6.0 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) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Weidong Wang, NRR
Hukam Garg, NRR
Ahsan Sallman, NRR

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3. Letter from Dietrich, Pete, ENO to USNRC, "Response to Request for Additional Information (RAI) Regarding Amendment Application for ARTS/MEOD Modification (TAC No. MC9681)," dated December 21, 2006.

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