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

June 8, 2009

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

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT RE: EMERGENCY LICENSE AMENDMENT REQUEST APPLICATION FOR TEMPORARY ONE-TIME CHANGES TO TECHNICAL SPECIFICATION 3.8.1, REQUIRED ACTION B.4 COMPLETION TIME (TAC NO. ME1404)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 294 to Renewed 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 June 4, 2009, as supplemented by the letter dated June 6, 2009.

The amendment authorizes a one-time change to TS 3.8.1 Required Action B.4 Completion Time. The amendment would add a note allowing a Completion Time of "17 days", on a one-time basis. This one-time allowance will expire at 10:15 a.m. on June 12, 2009.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice. Because this amendment was processed on an emergency basis, the *Federal Register* notice of issuance of the amendment will provide an opportunity for hearing.

Sincerely,

B.K. Vaidya

Bhalchandra K. Vaidya, 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. 294 to DPR-59
2. Safety Evaluation

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

ENTERGY NUCLEAR FITZPATRICK, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 294
Renewed Facility Operating 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 June 4, 2009, as supplemented on June 6, 2009, 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 Renewed 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. 294, 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 immediately.

FOR THE NUCLEAR REGULATORY COMMISSION

John P. Boska

John P. Boska, Acting Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: June 8, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 294

RENEWED 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
Page 3	Page 3

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contain marginal lines indicating the areas of change.

Remove Page	Insert Page
3.8.1-3	3.8.1-3

(4) ENO pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use, at any time, any byproduct, source and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration; or associated with radioactive apparatus, components or tools..

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; 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) Maximum Power Level

ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 2536 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A , as revised through Amendment No. 294, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

ENO shall implement and maintain in effect all provisions of the approved fire protections program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated November 20, 1972; the SER Supplement No. 1 dated February 1, 1973; the SER Supplement No. 2 dated October 4, 1974; the SER dated August 1, 1979; the SER Supplement dated October 3, 1980; the SER Supplement dated February 13, 1981; the NRC Letter dated February 24, 1981; Technical Specification Amendments 34 (dated January 31, 1978), 80 (dated May 22, 1984), 134 (dated July 19, 1989), 135 (dated September 5, 1989), 142 (dated October 23, 1989), 164 (dated August 10, 1990), 176 (dated January 16, 1992), 177 (dated February 10, 1992), 186 (dated February 19, 1993), 190 (dated June 29, 1993), 191 (dated July 7, 1993), 206 (dated February 28, 1994) and 214 (dated June 27, 1994); and NRC Exemptions and associated safety evaluations dated April 26, 1983, July 1, 1983, January 11, 1985, April 30, 1986, September 15, 1986 and September 10, 1992 subject to the following provision:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 Restore EDG subsystem to OPERABLE status.	14 days[1] <u>AND</u> 21 days from discovery of failure to meet LCO
C. Two offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when the redundant required feature(s) are inoperable. <u>AND</u> C.2 Restore one offsite circuit to OPERABLE status.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s) 7 days
D. One offsite circuit inoperable. <u>AND</u> One EDG subsystem inoperable	------- NOTE -------- Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems - Operating," when Condition D is entered with no AC power source to any division. ---------------- D.1 Restore Offsite circuit to OPERABLE status. <u>OR</u>	 12 hours (continued)

[1] For the "A" EDG subsystem only, the Completion Time that the subsystem can be inoperable as specified by Required Action B.4 may be extended beyond the "14 days AND 21 days from discovery of failure to meet LCO" up to "17 days AND 21 days from discovery of failure to meet LCO", to support repair and restoration of the 93EDG-C rotor. Upon Completion of the repair and restoration, this footnote is no longer applicable and will expire at 1015 on June 12, 2009.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 294 TO RENEWED 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 June 4, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML091550706), as supplemented by letter dated June 6, 2009, (ADAMS Accession No. ML091590066), Entergy Nuclear Operations, Inc. (the licensee) submitted a request for changes to the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Technical Specifications (TS). The supplement dated June 6, 2009, provided additional information that clarified the application and did not expand the scope of the application. Because this amendment was processed on an emergency basis, the *Federal Register* notice of issuance of the amendment will provide an opportunity for hearing.

The proposed change would allow, on a temporary one-time basis, the extension of the allowed outage time for the 'A' emergency diesel generator (EDG) subsystem from 14 days to 17 days. Specifically, the proposed amendment would revise the TS 3.8.1 Required Action B.4 Completion Time (CT), on a one-time basis by adding a footnote to the CT. The proposed note would read as follows:

> For the "A" Emergency Diesel Generator (EDG) subsystem only, the Completion Time that the subsystem can be inoperable as specified by Required Action B.4 may be extended beyond the "14 days and 21 days from discovery of failure to meet LCO [limiting condition for operation]" up to "17 days and 21 days from discovery of failure to meet LCO", to support repair and restoration of the 93EDG-C rotor. Upon completion of the repair and restoration, this footnote is no longer applicable and will expire at 1015 on June 12, 2009.

During the performance of the 2-year EDG preventive maintenance on EDG '93EDG-C', the licensee identified a deficiency with the EDG rotor. Through inspection and testing, the licensee has determined that one of the eight poles on the rotor must be rewound. After it identified the problem, the licensee transported the affected EDG rotor to an approved vendor facility for repair.

The licensee was concerned that completion of repairs, post-maintenance testing, and surveillance testing to reestablish operability may not be completed prior to expiration of the 14-day allowed outage time. Therefore, the licensee decided to request a one-time extension of this 14-day allowed outage time by an additional 3 days to assure adequate time is available for completion of repairs, post-maintenance testing, and surveillance testing of the EDG.

2.0 REGULATORY EVALUATION

The U.S. Nuclear Regulatory Commission (NRC) finds that the licensee in its June 4, 2009 submittal, as supplemented on June 6, 2009, identified the applicable regulatory requirements. The regulatory requirements and guidance which the NRC staff considered in its review of the application are discussed below.

2.1 Applicable Regulations

The following NRC requirements are applicable to the NRC staff's review of the licensee's amendment request:

Paragraph 50.36(c)(2)(ii) of Title 10 of the *Code of Federal Regulations* (10 CFR), "Technical specifications," requires that "[a] technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the [criteria set forth in 10 CFR 50.36(c)(2)(ii)(A)-(D)]."

While JAFNPP was not built or licensed to 10 CFR Appendix A of Part 50, General Design Criteria (GDC), it was evaluated and determined to meet the intent of Appendix A. As discussed below, GDC 17, "Electric power systems," requires two independent power sources. The NRC staff reviewed the proposed amendment and finds that the proposed amendment does not alter JAFNPP's compliance with the intent of GDC 17. The one-time allowance of a 17-day CT for TS 3.8.1 Required Action B.4 also does not change the requirement to restore the inoperable EDG to operable status.

GDC 17 requires, in part, that nuclear power plants have onsite and offsite electric power systems to permit the functioning of structures, systems, and components that are important to safety. The onsite system is required to have sufficient independence, redundancy, and testability to perform its safety function, assuming a single failure. The offsite power system is required to be supplied by two physically independent circuits that are designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. In addition, this criterion requires provisions to minimize the probability of losing electric power from the remaining electric power supplies as a result of loss of power from the unit, the offsite transmission network, or the onsite power supplies.

The principal design criteria for the electrical power system at JAFNPP are discussed in Updated Final Safety Analysis Report (UFSAR) Section 1.5.4, "Electrical Power System Criteria." The design criteria are as follows:

1. The Electrical Power System is designed to efficiently deliver the electrical power generated to transmission systems.

2. Sufficient normal, reserve and emergency sources of electrical power are provided to attain an orderly shutdown and to maintain the plant in a safe condition. The capacity of the power sources is adequate to accomplish all

required engineered safeguard functions under postulated design basis accident conditions.

The principal design criteria for emergency onsite electrical power at JAFNPP are discussed in UFSAR Section 1.5.6.4, "Emergency Onsite Power Criteria." The design criteria are as follows:

1. Emergency onsite electrical power sources are provided to allow orderly reactor shutdown and the removal of decay heat under circumstances where normal or reserve power is not available.

2. Emergency onsite electrical power sources have sufficient capacity to power the engineered safeguards required for safe shutdown after a design basis accident.

10 CFR 50.63, "Loss of all alternating current power," requires that each light-water cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout.

2.2 Applicable Regulatory Criteria/Guidelines

The regulatory guides (RG) on which the NRC staff based its acceptance are:

- RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (Ref. 5), describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

- RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," (Ref. 6), describes an acceptable risk-informed approach specifically for assessing proposed permanent TS changes in allowed outage times. This RG also provides risk acceptance guidelines for evaluating the results of such assessments. RG 1.177 identifies a three-tiered approach for the licensees evaluation of the risk associated with a proposed Completion Time (CT) TS change, as discussed below.

    - Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines that are consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174 and RG 1.177. The first tier assesses the impact on operational plant risk based on the change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF). It also evaluates plant risk while equipment covered by the proposed CT is out-of-service, as represented by incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP). Tier 1 also addresses probabilistic risk assessment (PRA) quality, including the technical adequacy of the licensee's plant-specific PRA for the subject application. Cumulative risk of the present TS change in light of past related applications or additional applications under review are also considered along with uncertainty/sensitivity analysis with respect to the assumptions related to the proposed TS change.

- Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, are taken out-of-service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The purpose of this evaluation is to ensure that there are appropriate restrictions in place such that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed CT is implemented.

- Tier 3 addresses the licensee's overall configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk-significant plant configurations resulting from maintenance or other operational activities and appropriate compensatory measures are taken to avoid risk significant configurations that may not have been considered when the Tier 2 evaluation was performed. Compared with Tier 2, Tier 3 provides additional coverage to ensure risk-significant plant equipment outage configurations are identified in a timely manner and that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule (10 CFR 50.65(a)(4)), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance testing and corrective and preventive maintenance, subject to the guidance provided in RG 1.177, Section 2.3.7.1, and the adequacy of the licensee's program and PRA model for this application. The CRMP is to ensure that equipment removed from service prior to or during the proposed extended CT will be appropriately assessed from a risk perspective.

  For temporary TS changes, examination of the risk metrics identified in RG 1.174 and RG 1.177 provides insight about the potential risk impacts, even though neither of these RGs provides numerical risk acceptance guidelines for evaluating temporary TS changes against the fourth key principle. It can be demonstrated with reasonable assurance that a temporary TS change meets the fourth key principle if the associated risk metrics:

  - Satisfy the risk acceptance guidelines in RG 1.174 and RG 1.177, or
  - Are not substantially above the risk acceptance guidelines in RG 1.174 and RG 1.177 and effective compensatory measures to maintain lower risk are implemented while the temporary TS change is in effect.

- RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (Ref. 7), describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light water-reactors.

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in Chapter 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," of the NRC Standard Review Plan (SRP), NUREG-0800 (Ref. 6). Guidance on evaluating PRA technical adequacy is provided in Chapter 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Ref. 7). More specific guidance related to risk-informed TS changes is provided in SRP Section 16.1, "Risk-Informed Decisionmaking: Technical Specifications," (Ref. 8), which includes CT changes as part of risk-informed decision making. Chapter 19.2 of the SRP states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

- The proposed change meets the current regulations, unless it is explicitly related to a requested exemption..
- The proposed change is consistent with the defense-in-depth philosophy.
- The proposed change maintains sufficient safety margins.
- When proposed changes increase core damage frequency or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.
- The impact of the proposed change should be monitored using performance measurement strategies.

For temporary changes, SRP Chapters 19.2 and 16.1 are used to provide general guidance regarding evaluation of the potential risk impacts.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's analysis in support of its proposed license amendment, which are described in the original submittal dated June 4, 2009 (Ref. 1), as supplemented by letter dated June 6, 2009 (Ref. 2).

3.1 Review Methodology

Per SRP Chapter 19 and Section 16.1, the NRC staff reviewed the submittal using the three-tiered approach and the five key principles of risk-informed decision making presented in RG 1.174 and RG 1.177.

3.2 Key Information Used in the Review

The key information used in the NRC staff's review is contained Section 4.2 of the enclosure of Reference 1 including referenced Attachments 4 and 5, and Reference 2.

3.3 Comparison Against Regulatory Criteria/Guidelines

The NRC staff's evaluation of the licensee's proposed changes to TS 3.8.1, "Alternating Current (AC) Sources," uses the three-tiered approach and the five key principles outlined in RGs 1.174 and 1.177.

3.3.1 Deterministic Engineering Evaluation

The deterministic evaluation addresses key principles 1, 2, 3, and 5 of the NRC staff's philosophy of risk-informed decision making, which concerns compliance with current regulations, evaluation of defense-in-depth, evaluation of safety margins, and performance monitoring strategies.

The JAFNPP emergency power system consists of four EDGs each located in a separate room within the EDG building, connected to the "A" and "B" emergency buses to supply emergency power during a loss of offsite power event. Each of the two independent and redundant emergency power systems (i.e., divisions) consists of an EDG pair connected to emergency switchgear, which contains the emergency bus, generator output and tie circuit breakers, and the emergency core cooling system (ECCS) load circuit breakers. The EDGs are designed to provide an alternate, onsite source of reliable 4160 volt (V) alternating current (AC) power for safe shutdown equipment required to mitigate the consequences of a design-basis accident (DBA) in the event of a total loss of the normal and offsite power sources. Each generator has a continuous rating of 2600 kilo-watts (kW); therefore, the total loading capacity available per division, with both EDGs in the divisional pair operating, is 5200 kW at 4160 V AC and 60 hertz (Hz). Each EDG also has short time rating of 2850 kW for 2000 hours, 2950 kW for 160 hours and 3050 kW for 30 minutes.

The worst case automatic loading (normal and emergency) for the "A" emergency bus with a single EDG supplying power is 3179.1 kW, which excludes the second residual heat removal (RHR) pump that is blocked from starting if one EDG in a divisional pair fails to start. Operators can manually start the blocked RHR pump as needed within the EDG capacity, as directed by emergency, abnormal, and normal operating procedures. The RHR system would be capable of providing the 100% capacity divisional function that is required for the RHR system to perform the low pressure injection function with a single operating EDG in the division. In the current configuration with 93EDG-C out-of-service, the licensee has transferred non-vital loads to the "B" emergency bus in order to reduce the worst case automatic loading on the "A" emergency bus to 2964.2 kW, within the short time capacity rating of the remaining 93EDG-A.

Abnormal operating procedures (AOPs) address the loss of individual 4160 V AC buses, the loss of station batteries, and in the worst case, station blackout. The licensee stated that these procedures are periodically reviewed in licensed and non-licensed operator continuing training. These procedures provide guidance for achieving a safe shutdown condition.

In addition to the AOPs, JAFNPP has a strategy of extending the station blackout (SBO) coping time. Guidance for this strategy is provided in Technical Support Guideline (TSG)-8, "Extending Station Blackout Time." TSG-8 provides direction to start the EDG manually without electrical power available, flashing the field if the EDG does not self-excite, and ensure cooling water supply. Reactor core isolation cooling (RCIC) operation time is extended by providing AC power to a station battery charger using a portable generator. In addition, instructions are provided to manually operate RCIC with no direct current power available. As an added measure of assurance, the licensee stated that all necessary equipment will be pre-staged.

The licensee provided a Regulatory Commitment to stage a 1500 kW, 4160 V temporary diesel generator (DG) on-site as a back-up power supply. This power supply will be available to be connected to a vital bus in the event of an SBO, should the plant abnormal operating procedure strategies for restoring power be unsuccessful. The NRC staff understands that the licensee will start the temporary DG once it has been staged and then periodically (every 8 hours) verify that the generator is properly staged and that the guidance necessary to connect it to the emergency bus is available at the machine. The licensee will ensure that appropriate guidance for using this equipment will be in place prior to entering the extended allowed outage time period. The NRC staff requested the licensee to provide the basis for selecting a 1500 kW DG. In response to the NRC staff's request, the licensee stated that the temporary DG was selected to provide a pre-staged source of AC power that can be connected to one of the station's emergency buses to allow mitigation of an SBO event following a total loss of offsite and onsite AC power. Based on this response, the NRC staff understands that the temporary 1500 kW DG is sufficiently sized to supply adequate power to mitigate the consequences of a station blackout event. The licensee stated that this power source is capable of powering a 125 V battery charger, an RHR pump along with an RHR service water pump and sufficient 600 V loads to support their operation. This complement of pumps is sufficient to provide core and containment cooling in the absence of large break loss-of-coolant-accidents affecting the reactor water recirculation system. Connection of the 1500 kW DG to an emergency bus effectively terminates the SBO by providing AC power for the station. The licensee further stated that instructions for connecting the power source are contained with Technical Support Guidelines that support the Severe Accident Operating Guidelines. The NRC staff also requested the licensee to provide the specific requirements for this temporary DG (e.g., time and actions required for the temporary DG to power the safety bus). In response to the NRC staff's request, the licensee stated that connection of the temporary DG could be accomplished within the 4-hour station blackout coping period. The licensee stated and the NRC staff agrees that it has developed procedures for connecting the temporary DG to the safety bus and trained the operations staff on those procedures. In addition, the licensee stated that a qualified JAFNPP operator and electrician walked-through these procedures and estimated the time to energize the safety bus from the temporary DG is approximately 3 hours, which is within the 4-hour SBO coping time for JAFNPP. In response to an NRC staff request for additional information (RAI), the licensee provided reasonable assurance that adequate fuel oil will be available for the temporary DG to operate continuously at rated load for the duration of the extended allowed outage time.

In its submittal, the licensee stated that it has reviewed the historical preventive maintenance test data for the other EDGs (i.e., 93EDG-A, 93EDG-B and 93-EDG-D) and determined that the deficiency does not extend to those EDGs. In an RAI, the NRC staff requested the licensee to provide a detailed discussion on how common mode failure was ruled out. In response to this RAI, the licensee provided a detailed discussion and the historical preventive maintenance test data. The NRC staff reviewed this information and finds that the licensee's assessment provides reasonable assurance that common mode failure should not be an issue for the other EDGs.

Conclusion – Deterministic Evaluation

Based on the above evaluation, the NRC staff finds the proposed revision to the JAFNPP TSs will continue to ensure the availability of the required AC power to shut down the reactor and to maintain the reactor in a safe condition after an anticipated operational occurrence or a postulated DBA. The NRC staff also concludes that the proposed TS change does not alter JAFNPP's compliance with the intent of GDC 17 or the requirements of 10 CFR 50.36 and 10

CFR 50.63. The NRC staff further concludes that the change will not alter JAFNPP's compliance with the principal design criteria in UFSAR Sections 1.5.4 and 1.5.6.4 since the change will not impact the redundancy or availability requirements of offsite power supplies or change the ability of the plant to cope with an SBO event. The NRC staff's conclusion is also based on the licensee implementing the Regulatory Commitments listed in Section 3.4 of this safety evaluation. Based on this information, the NRC staff finds the proposed change acceptable.

3.3.2 NRC Staff Probabilistic Risk Assessment (PRA) Evaluation

The evaluation presented below addresses the NRC staff's philosophy of risk-informed decision making, that when the proposed changes result in a change in CDF or risk, the increase should be small and consistent with the intent of the Commission's Safety Goal Policy Statement (Key Principle 4).

3.3.2.1 Tier 1: PRA Capability and Insights

The first tier evaluates the impact of the proposed changes on plant operational risk. The Tier 1 NRC staff review involves two aspects: (1) evaluation of the validity of the JAFNPP PRA models and their application to the proposed changes, and (2) evaluation of the PRA results and insights based on the licensee's proposed application.

PRA Quality – Internal Events Model

The objective of the PRA quality review is to determine whether the JAFNPP PRA used in evaluating the proposed change to TS 3.8.1 CT is of sufficient scope, level of detail, and technical adequacy for this application. The NRC staff review evaluated the PRA quality information provided by the licensee in their submittals, including industry peer review results and self-assessments performed by the licensee.

The JAFNPP PRA model addresses both CDF and LERF for internal events, at-power conditions. The model is based on the original model developed to support the Individual Plant Examination (IPE). Since 1991, updates have been made to incorporate plant design and procedure changes, update plant-specific reliability and unavailability data, and to improve the fidelity of the model. The model is maintained through a periodic review and update process. Of specific relevance to this application, the initiating event frequencies and component failure data were updated based on NUREG/CR-6928, "Industry Average Performance for Components and Initiating Events at U. S. Commercial Nuclear Power Plants", February 2007, and the offsite power recovery model was updated based on NUREG/CR-6890, "Reevaluation of Station Blackout Risk at Nuclear Power Plants Analysis of Loss of Offsite Power Events: 1986 – 2004," December 2005, supplemented by more recent industry data through December 2007.

In 1997, the Boiling Water Reactor Owners Group (BWROG) performed a peer review of the JAFNPP PRA model. All peer review identified issues and observations have been addressed and incorporated into the PRA model. The licensee stated that it has performed a "gap assessment" using the internal events PRA standard (American Society of Mechanical Engineers, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME-RA-Sb-2005) and the guidance of RG 1.200. The assessment was actually performed for other similar plant models developed by the licensee PRA staff which use the same PRA methods employed at JAFNPP. Based on these commonalities, the licensee stated that the gap assessment performed for these PRA models would directly apply to JAFNPP. In addition, the licensee has incorporated changes to the JAFNPP PRA model which address the gap assessment issues. The licensee identified findings from its internal review when compared to the capability category II supporting requirements of the standard, and provided its disposition of these items for the application of an extended EDG CT. The NRC staff reviewed the licensee's disposition of the gap assessment issues for this application, as follows:

- Eight items involve documentation issues of sources of uncertainty in the PRA model. Areas of model uncertainty for this application have been identified and evaluated in the amendment request. Therefore, the lack of documentation does not impact the application.
- Two items involve the conservative treatment of harsh environment for equipment due to plant conditions after core damage. The licensee determined that the physical location of the Emergency Diesel Generators (EDGs) would not involve any harsh environmental conditions following a core damage event, and so these standard elements are not relevant to this application.
- Six items involve model issues associated with LERF. The licensee determined that the unavailability of a single EDG is not a significant contributor to increases in LERF, and therefore these standard elements are not significant to this application.
- One item involves documentation associated with LERF models. The licensee determined that this documentation issue has no impact on the application, and so the standard element is not relevant to this application.

Based on the licensee's dispositions above, the NRC staff agrees that the issues identified with nonconformance to capability category II of the internal events standard would not have a significant impact on the risk assessment results supporting this request.

The licensee stated that the JAFNPP PRA model includes plant configuration and procedure changes through September 2008, and that the modifications and procedure changes since this date have been reviewed and do not adversely affect the JAFNPP PRA model. Therefore, the PRA model reasonably reflects the as-built as-operated facility.

The loss of off site power (LOOP) initiating event frequency for the JAFNPP PRA model is 4.3E-2/year, which reflects industry data from 1997 – 2007. In addition, the model includes a consequential LOOP after a plant trip. The probability of recovery of offsite power uses industry time duration data through 2007. The licensee has concluded that, based on a review of preventive maintenance test data, a similar fault condition does not exist in any of the three remaining operable EDGs. Therefore, no increased common cause failure probability is necessary for the risk evaluation. The licensee also confirmed that the PRA model does not credit any repairs of the EDGs, which is reasonable and conservative for this application.

Therefore, based on the above information, the NRC staff finds that the licensee has satisfied the intent of RG 1.177 (Sections 2.3.1, 2.3.2, and 2.3.3), RG 1.174 (Section 2.2.3 and 2.5), RG 1.200, and SRP Chapter 19.1, and that the quality of the JAFNPP internal events PRA is sufficient to support the risk evaluation provided by the licensee in support of the proposed license amendment.

PRA Quality – Internal Fires Model

The licensee provided a separate quantitative estimate of the impact of the EDG outage on fire risk. The fire risk was evaluated using a fire PRA model from the JAFNPP Individual Plant Examination of External Events (IPEEE), which was developed using the guidance available from the Electric Power Research Institute (EPRI) Fire PRA Implementation Guide. The licensee stated that this PRA model is conservative and identified specific plant improvements made since the IPEEE but not included in the fire PRA model, as well as conservative assumptions related to the data used in the fire PRA model. The model has not been updated since the IPEEE (circa 1996). However, the licensee stated that plant changes made since the IPEEE was completed would result in risk reductions, which would tend to reduce the fire risk.

Based on the conservative assumptions identified, and on the licensee's evaluation that the changes in the plant design and operation would tend to reduce risk associated with fires, the staff finds that the licensee has satisfied the intent of RG 1.177 (Sections 2.3.1, 2.3.2, and 2.3.3), RG 1.174 (Section 2.2.3 and 2.5), RG 1.200, and SRP Chapter 19.1, and that the quality of the fire PRA and methods applied is sufficient to support the risk evaluation provided by the licensee in support of the proposed license amendment.

PRA Risk Results and Insights

The licensee determined the configuration-specific risk associated with the inoperable 'C' EDG. Unavailability of other plant equipment at their nominal average values was assumed. The licensee used an average unavailability assumption for other modeled equipment. In addition, no credit was taken in the PRA for the risk management actions identified by the licensee to be implemented during the extended EDG outage. The ICCDP and ICLERP are based on the entire 17-day duration of the proposed extended CT.

The licensee's methodology is consistent with the guidance of RG 1.177, Section 2.3.4 and Section 2.4 and is, therefore, acceptable to the NRC staff.

The results are as follows:

Risk Measure	Internal Events	Fire Events
ICCDP	5.82E-8	4.36E-8
ICLERP	2.98E-9	Not Provided

The licensee did not provide an estimate of the ICLERP associated with fire events. The NRC staff notes that even if it is conservatively assumed that 100% of the fire events leading to increased core damage probability also result in large early releases, the total ICLERP would still be below the RG 1.177 guidance for permanent TS changes.

The licensee identified the areas in the plant which dominate the fire risk results. Additional risk management actions are identified to compensate for the increased risk (discussed below for Tier 2).

The licensee did not explicitly provide an estimate of the ΔCDF and ΔLERF associated with this proposed change. The ΔCDF and ΔLERF may be determined by assuming a frequency for entry into an extended CT of this nature. Because the proposed TS change is a temporary change to be implemented one time only during the current calendar year, it can be conservatively assumed that the frequency of the extended CT is 1/year, and so the ΔCDF and ΔLERF are numerically identical with the ICCDP and ICLERP.

Per RG 1.177, the acceptance guidelines for ICCDP and ICLERP for permanent TS changes are 5E-7 and 5E-8, respectively. Per RG 1.174, the acceptance guidelines for ΔCDF and ΔLERF are 1E-6/year and 1E-7/year, respectively, for very small changes in risk. The licensee's estimates are consistent with these guidelines applicable to permanent changes, and are therefore considered acceptable to permit a temporary one-time change.

Qualitative Evaluation of Seismic Risk

The licensee did not quantitatively assess the impact of seismic events on CDF or LERF, but instead provided a qualitative assessment of the impact of seismic events during the transformer outage. The seismic margins assessment, performed for the IPEEE, identified that a station blackout event with seismic failure of the EDG building block walls is the dominant contributor to risk. Since the availability status of the EDGs is not relevant to this dominant seismic sequence, the licensee has concluded that the unavailability of the 'C' EDG would not have any significant impact on seismic risk.

The NRC staff finds that the licensee has provided sufficient information to support this risk evaluation.

Qualitative Evaluation of Other External Events

The licensee reviewed its IPEEE evaluation of other external hazards, including high winds and tornadoes, severe weather, lightning, external flooding, aircraft impacts, and transportation and nearby facility accidents. No risk-significant external hazards were identified, and therefore the licensee concluded that the unavailability of the 'C' EDG would not have any significant impact on the risk from such events.

The NRC staff finds that the licensee has provided sufficient information to support this risk evaluation.

Shutdown Risk

The licensee's submittal did not discuss shutdown risk, since the proposed change is only applicable during power operation. This is conservative, since the licensee is avoiding additional shutdown risk by completing EDG repairs during operation, but the risk calculations do not consider this avoided risk.

Uncertainty Analysis

The licensee identified the LOOP initiating event frequency and the EDG common cause failure probability as potential sources of uncertainty which could impact the risk results. In order to quantitatively assess the potential impact, two sensitivity cases were evaluated which doubled the LOOP initiator frequency and doubled the EDG common cause failure probability. The results demonstrated a nearly linear increase in the change in CDF, which is expected since the risk results are dominated by LOOP and failure of the remaining EDGs. The NRC staff finds that the sensitivity analyses performed demonstrate that the results are in fact sensitive to uncertainties in LOOP frequency and EDG failure probabilities. However, because there is significant margin available between the calculated risk impact and the acceptance guidance, uncertainties in these parameters would not change the regulatory decision.

3.3.2.2 Tier 2: Avoidance of Risk-Significant Plant Configuration

The licensee has identified plant equipment which will be administratively protected and not voluntarily removed from service for any routine work activities during the extended EDG outage. Further, the existing plant TS control the availability of the remaining train EDGs and the two offsite power circuits.

The licensee has also identified additional risk management actions related to the important fire areas based on the fire risk evaluation. These are increased administrative control over hot work activities in the vicinity of protected equipment and in the impacted fire zones, no planned maintenance on fire detection or suppression equipment in the impacted fire zones, and the removal of any unnecessary transient combustibles in the impacted fire zones.

The licensee is also staging a temporary diesel generator on site as a backup power supply for a vital bus, to provide for additional mitigation capability of a station blackout event. No credit has been taken in the risk analyses for this diesel generator.

3.3.2.3 Tier 3: Risk-Informed Configuration Risk Management

The licensee stated that its 10 CFR 50.65(a)(4) program for at-power risk management requires a risk assessment prior to performing maintenance activities. The licensee's program conforms to the guidance of NUMARC 93-01 and RG 1.182, and therefore is considered to satisfy the intent of RG 1.177 for a configuration risk management program.

3.4.3 Conclusion – PRA Evaluation

The risk impact of the proposed one-time 17-day completion time for the completion of repairs to the 'C' EDG is consistent with the acceptance guidelines specified in RG 1.174, RG 1.177, and NRC staff guidance outlined in Chapter 16.1, "Risk-Informed Decisionmaking: Technical Specifications," of NUREG-0800. The Tier 2 evaluation identified the applicable risk-significant plant equipment outage configurations needing compensatory measures that will be implemented by the licensee during the extended EDG outage. The licensee's 10 CFR 50.65(a)(4) program satisfies the CRMP guidance in RG 1.177. Therefore, the NRC staff finds that the risk analysis methodology and approach used by the licensee to estimate the risk impacts and manage configuration risk during the extended transformer outage are reasonable and of sufficient quality.

Based on the above, the NRC staff finds the proposed one-time change to revise the Completion Time of Required Actions of TS 3.8.1, associated with one inoperable EDG train, to be acceptable.

3.4 List Of Regulatory Commitments

The licensee, in its application dated June 4, 2009, as supplemented by letter dated June 6, 2009, included regulatory commitments. The commitments are listed in the following table.

<table>
<tr><th rowspan="2">COMMITMENT</th><th colspan="2">TYPE
(Check one)</th><th rowspan="2">SCHEDULED COMPLETION DATE
(If Required)</th></tr>
<tr><th>ONE-TIME ACTION</th><th>CONTINUING COMPLIANCE</th></tr>
<tr><td>The following equipment will be protected in accordance with the plant Protected Equipment Program AP-12.12, during the period of extended AOT for 93EDG-C. The Protected Equipment Program requirements include 1) posting the equipment with signs and barriers to prevent inadvertent operation; 2) no routine / elective work activities on protected equipment, this list of equipment will not be voluntarily removed from service; and 3) Operations Manager approval for any emergent work involving protected equipment.

o Emergency Diesel Generators 93EDG-A, 93EDG-B and 93EDG-D
o Emergency Service Water Pumps 46P-2A and 46P-2B
o 4160V Normal and Emergency Switchgear Buses 10300, 10400, 10500 and 10600
o Station Batteries 71SB-1 and 71SB-2
o Station Battery Chargers 71BC-1 and 71BC-2
o 125-Vdc Control boards 71BCB-2A and 71BCB-2B
o Main Transformers 71T-1A, and 71T-1B
o Normal Station Service Transformer 71T-4
o Reserve Station Service Transformers 71T-2, and 71T-3
o North and South 115 kV Bus Reserve Station Service Transformer Disconnect Switches 71EDSC-10015, 71EDSC-10017, and 71EDSC-10025
o RHR/RHRSW Loops "A" & "B"
o HPCI pump 23P-1
o RCIC pump 13P-1
o Torus vent valves 27AOV-117 and 27AOV-118
o Diesel Driven Fire Pump 76P-1
o Diesel Driven Fire Pump 76P-4
o 115 kV Switchyard</td><td>X</td><td></td><td>Prior to entering the period of extended AOT and maintained for the duration of the extended AOT.</td></tr>
</table>

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Transfer non-vital loads from the "A" emergency bus to the "B" emergency bus to reduce the "A" bus loading to within the short time capacity of 93EDG-A.	X		Prior to entering the period of extended AOT and maintained for the duration of the extended AOT.
Stage a 1500 kW, 4160v temporary diesel generator on-site as a back-up power supply. This power supply will be available to be connected to a vital bus in the event of a Station Blackout, should the plant AOP strategies for restoring power be unsuccessful. Appropriate guidance for using this equipment will be in place prior to entering the extended AOT period.	X		Prior to entering the period of extended AOT and maintained for the duration of the extended AOT.
Increased administrative control will be exercised for any proposed hot work in the vicinity of protected equipment and in the impacted fire zones (CT-2 (East Cable Tunnel), EG-6 (Emergency Diesel Switchgear Room), and BR-4 (Train B Battery Charger Room)).	X		Prior to entering the period of extended AOT and maintained for the duration of the extended AOT.
No planned maintenance on fire detection or fire suppression equipment that will cause the fire detection or fire suppression equipment in the impacted fire zones (CT-2 (East Cable Tunnel), EG-6 (Emergency Diesel Switchgear Room), and BR-4 (Train B Battery Charger Room)) to be inoperable.	X		For the duration of the extended AOT.
Transient combustible loading in these areas (CT-2 (East Cable Tunnel), EG-6 (Emergency Diesel Switchgear Room), and BR-4 (Train B Battery Charger Room) EDG Rooms) will be reviewed and any unnecessary transient combustibles will be removed.	X		Prior to entering the period of extended AOT and maintained for the duration of the extended AOT.
If an equipment failure occurs that affects the protected equipment noted above, the applicable Technical Specification Conditions will be entered, and Senior Plant management will be notified.	X		Prior to entering the period of extended AOT and maintained for the duration of the extended AOT.
Maintenance and surveillance activities which could lead to Main Turbine trip will be avoided.	X		For the duration of the extended AOT.
The plant Operations crew and Maintenance staff will be briefed on these risk management measures.	X		Prior to entering the period of extended AOT and maintained for the duration of the extended AOT.

COMMITMENT	TYPE (Check one) ONE-TIME ACTION	CONTINUING COMPLIANCE	SCHEDULED COMPLETION DATE (If Required)
The system load dispatcher will be contacted once per day to ensure no significant grid perturbations are expected during the extended AOT. Also, the system load dispatcher should inform the plant operator if conditions change during the extended AOT (e.g., when the predicted voltages would be unacceptable as a result of a trip of the nuclear unit).	X		For the duration of the extended AOT.
Just-in-time training will be provided to the operating shifts to heighten their awareness of challenges to the electrical distribution system in this configuration. This will include review of electrical distribution related AOPs, AOP-28, TSG-8, and the guidance associated with the temporary diesel generator staged as a compensatory measure.	X		Prior to entering the period of extended AOT and prior to any individual assuming the shift.
Operations will monitor weather conditions to assess potential impacts on plant conditions due to adverse weather conditions.	X		For the duration of the extended AOT
These compensatory measures will be promulgated to the operating crews in an operations department standing order.	X		Prior to entering the period of extended AOT.
The operations department will periodically (every 8 hours) verify that the generator is properly staged and that the guidance necessary to connect it to the emergency bus is available at the machine.	X		Prior to entering the period of extended AOT and maintained for the duration of the extended AOT.

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitments are provided by the licensee's administrative processes, including its commitment management program. Should the licensee choose to incorporate a regulatory commitment into the emergency plan, Updated Final Safety Analysis Report (UFSAR), or other documents with established regulatory controls, the associated regulations would define the appropriate change-control and reporting requirements. The NRC staff has determined that the commitments do not warrant the creation of regulatory requirements, which would require prior NRC approval of subsequent changes. The NRC staff has agreed that Nuclear Energy Institute 99-04, Revision 0, "Guidelines for Managing NRC Commitment Changes," provides reasonable guidance for the control of regulatory commitments made to the NRC staff (see Regulatory Issue Summary 2000-17, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff," dated September 21, 2000). The commitments should be controlled in accordance with industry guidance or comparable criteria employed by a specific licensee. The NRC staff may choose to verify the implementation and maintenance of these commitments in a future inspection or audit.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92(c) state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or,

(2) Create the possibility of a new or different kind of accident from any previously evaluated; or,

(3) Involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), in its June 4, 2009, application as supplemented on June 6, 2009, the licensee provided the analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

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

   Response: No.

   The proposed license amendment introduces a one-time 17 day completion time allowance for TS 3.8.1, Required Action B.4. The proposed completion time does not introduce any new accident initiators. The probability of an accident occurring is not affected by the proposed completion time. The consequences of the accidents evaluated in the UFSAR Accident Analysis in terms of delta CDF, ICCDF, and ICLERP remain within the thresholds identified in Regulatory Guide 1.77.

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

   Response: No.

   The proposed amendment makes a one-time allowance of a 17 day completion time for TS 3.8.1 Required Action B.4. The proposed amendment does not introduce any new equipment, create any new failure modes for existing equipment, or create any new limiting single failures. The plant equipment considered when evaluating the existing completion time remains unchanged. The temporary diesel staged as a compensatory measure is not considered to be new equipment since it would only be connected to the plant after an accident or transient had already occurred. The extended completion time will permit completion of repair activities without incurring transient risks associated with performing a shutdown with one EDG unavailable.

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

   Response: No.

   The proposed license amendment makes a one-time allowance of a 17 day completion time for TS 3.8.1 Required Action B.4. The current completion time includes an allowance of "21 days from discovery of failure to meet LCO [limiting condition for operation]". While this allowance is provided to account for overlapping LCO Conditions involving multiple trains it is indicative that operation with a single train of emergency power for 21 days has been reviewed and found to be acceptable. The proposed completion time has been evaluated using the JAFNPP PRA Model as discussed above. The use of a one-time completion time of 17 days results in an ICCDP of 5.82E-08 which is below the ICCDP guidance of 5E-07, and an ICLERP of 2.98E-09 which is below the ICLERP guidance of 5E-08. Therefore the proposed amendment does not involve a significant reduction in any margin of safety.

Based on the above discussion, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that the amendment request involves no significant hazards consideration.

5.0 EMERGENCY CIRCUMSTANCES

In its June 4, 2009, application as supplemented on June 6, 2009, the licensee requested that this amendment be treated under emergency circumstances. In accordance with 10 CFR 50.91(a)(5), the licensee provided the following justification regarding why this emergency situation occurred and how it could not be avoided:

> In accordance with 10 CFR 50.91(a)(5) Entergy has requested a one-time allowance to increase a Technical Specification LCO 3.8.1 Required Action B.4 from "14 days AND 21 days from discovery of failure to meet LCO" to "17 days AND 21 days from discovery of failure to meet LCO."
>
> LCO 3.8.1 Condition B was entered at 1015 on May 26, 2009, in order to perform scheduled maintenance on Emergency Diesel Generator (EDG) 93EDG-C. This EDG is paired with 93EDG-A to form one of two redundant EDG subsystems. At the time the LCO Condition was entered there was no indication of problem with 93EDG-C. The surveillance tests to demonstrate Operability were current and the last performance of the insulation resistance test (meggering) Preventive Maintenance (PM) task had results that were in the acceptable range with adequate margin.
>
> At approximately 0100 on May 28, 2009, a low megger reading was identified during performance of the 2-year PM to perform insulation resistance testing on the 93EDG-C rotor. At that time the issue was entered into the corrective action program and efforts to define the problem and the scope of repair commenced.

With support from vendor personnel the 93EDG-C rotor was removed on May 29, 2009 and inspected. Based on the results of the inspection the rotor was transported to the vendor's repair facility for further testing and inspection. As a result of the activities at the vendor's facility on May 31, 2009 it was determined that one of the eight poles on the rotor was shorted and would have to be rewound. Once the scope of the repair was understood a preliminary schedule was prepared that indicated completion of the repair activity might challenge the ability of the plant staff to complete the restoration and testing within the 14 day completion time allowed by the LCO. On June 2, 2009 a draft amendment request was discussed with NRC Staff and a submittal formally requesting the extension was filed on June 4, 2009. During this time work on the repair of the 93EDG-C rotor has continued on a round-the-clock basis.

Since Entergy had no reason to expect an emergent failure of the rotor, no Technical Specification amendment was requested in advance. As a consequence neither the routine amendment processing time, which includes a 30 day public comment period, nor the exigent amendment processing time, which includes a 14 day public comment period, support a decision prior to the expiration of the current Required Action B.4 Completion Time, on June 9, 2009.

Based on this review Entergy believes that the criteria in 10 CFR 50.91(a)(5) regarding an emergency that could not be avoided by the licensee is met and that processing the requested amendment on an emergency basis is warranted.

The NRC staff has reviewed the licensee's chronology of events leading to the emergency circumstances. The NRC staff finds that an emergency situation exists, in that failure to act in a timely way would have resulted in the required shutdown of JAFNPP. Based on the licensee's explanation, the NRC staff finds that the licensee could not have filed this application for a license amendment within the normal time frame. The NRC staff has presented its final no significant hazards consideration determination. Accordingly, the NRC staff finds that the requirements in 10 CFR 50.91(a)(5) have been met and that the licensee has not abused the emergency provision of this regulation. The requested amendment can be issued without prior notice. An opportunity for a hearing will be published in the *Federal Register* with the notice of issuance.

6.0 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.

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 NRC staff has presented its final no significant hazards consideration determination in Section 4.0 of this Safety Evaluation. The amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 10 CFR 51.22(c)(10). 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 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.

9.0 REFERENCES

1. Letter, P. Dietrich to U. S. Nuclear Regulatory Commission, "Emergency License Amendment Request Application for Technical Specification 3.8.1. Required Action B.4 Completion Time," June 4, 2009.
2. Letter, P. Dietrich to U. S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding: Emergency License Amendment Request Application for Technical Specification 3.8.1. Required Action B.4 Completion Time (TAC No. ME1404)," June 6, 2009.
3. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," USNRC, Revision 1, November 2002.
4. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," USNRC, August 1998.
5. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," USNRC, Revision 2, March 2009.
6. NUREG-0800, Standard Review Plan 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," Revision 0, June 2007.
7. NUREG-0800, Standard Review Plan 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, June 2007.
8. NUREG-0800, Standard Review Plan 16.1, "Risk-Informed Decisionmaking: Technical Specifications," Revision 1, March 2007.

Principal Contributors: M. McConnell
A. Howe

Date: June 8, 2009

June 8, 2009

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

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT RE: EMERGENCY LICENSE AMENDMENT REQUEST APPLICATION FOR TEMPORARY ONE-TIME CHANGES TO TECHNICAL SPECIFICATION 3.8.1, REQUIRED ACTION B.4 COMPLETION TIME (TAC NO. ME1404)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 294 to Renewed 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 June 4, 2009, as supplemented by the letter dated June 6, 2009.

The amendment authorizes a one-time change to TS 3.8.1 Required Action B.4 Completion Time. The amendment would add a note allowing a Completion Time of "17 days", on a one-time basis. This one-time allowance will expire at 10:15 a.m. on June 12, 2009.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice. Because this amendment was processed on an emergency basis, the *Federal Register* notice of issuance of the amendment will provide an opportunity for hearing.

Sincerely,
/ra/

Bhalchandra K. Vaidya, 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. 294 to DPR-59
2. Safety Evaluation
cc w/encls: Distribution via Listserv

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ADAMS Accession No.: ML091550348 *No substantial changes to SE Input Memo

OFFICE	LPL1-1\PM	LPL1-1\LA	NRR/EEEB\BC	NRR/APLA\BC	NRR/ITSB\BC	OGC	LPL1-1\BC
NAME	B. K. Vaidya	SLittle	GWilson*	DHarrison	RElliot	MSmith	JBoska
DATE	6/8/09	6/8/09	06/08/09	6/8/09 w/ changes	6/8/09	6/8/09	6/8/09

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