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June 4, 2009
JAFP-09-0070

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

SUBJECT: Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
License No. DPR-59
Emergency License Amendment Request Application for Technical Specification 3.8.1 Required Action B.4 Completion Time

REFERENCE: Technical Specification 3.8.1, AC Sources Operating

Dear Sir or Madam:

Pursuant to 10 CFR 50.90 and 10 CFR 50.91(a)(5), Entergy Nuclear Operations, Inc, (Entergy) requests Nuclear Regulatory Commission (NRC) review and approval of a proposed emergency license amendment for the James A. FitzPatrick Nuclear Power Plant (JAF). Entergy proposes a one-time change to Technical Specification (TS) 3.8.1 Required Action B.4 Completion Time. This request is to add a note allowing a Completion Time of "17 days", on a one-time basis. This one-time allowance will expire at 1015 on June 12, 2009.

During the performance of the 2-Year Emergency Diesel Generator (EDG) Preventive Maintenance (PM) a deficiency with the rotor on 93EDG-C was identified. Through inspection and testing it has been determined that one of the eight poles on the rotor must be rewound. Review of test data determined that the deficiency does not extend to JAF's other three safety-related EDGs. The proposed change is required to complete rewind of the rotor pole and return the EDG to operable status without requiring a plant shutdown.

Attachment 1 provides a description and evaluation of the proposed TS changes.
Attachment 2 provides the proposed changes to the current TS on marked up pages.
Attachment 3 provides the proposed TS changes in final typed format.
Attachment 4 provides simplified diagrams of the Electrical Distribution System

Entergy requests approval of the proposed License Amendment by June 8, 2009, with the amendment being implemented immediately.

In accordance with 10 CFR 50.91, a copy of this application, with the associated attachments, is being provided to the designated New York State official.

There are no commitments contained in this letter. Should you have any questions concerning this submittal, please contact Mr. Joseph Pechacek at 315-349-6766. Entergy will supplement this request as necessary to resolve NRC staff questions.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on the 4th day of June, 2009.

Sincerely,

Pete Dietrich
Site Vice President

PD/JP/ed

Attachments: 1. Description and evaluation of the proposed TS changes
2. Proposed changes to the current TS on marked-up pages
3. Proposed TS changes in final typed format
4. Simplified Electrical Distribution Diagrams

cc: next page

cc:

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JAFP-09-0070
Attachment 1

Description and Evaluation

Emergency License Amendment Request Application for Technical Specification 3.8.1 Required Action B.4 Completion Time

1.0 Description

The proposed amendment would revise the Technical Specification (TS) 3.8.1 Required Action B.4 Completion Time, on a one-time basis by adding a footnote to the Completion Time. The proposed note would read "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" 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 (PM), a deficiency with the rotor on 93EDG-C was identified. Through inspection and testing it, has been determined that one of the eight poles on the rotor must be rewound. The affected EDG rotor has been transported to an approved vendor facility where repairs are in progress. James A. FitzPatrick Nuclear Power Plant (JAF) personnel are on-site at the repair facility providing continuous oversight of the repair activity.

Review of PM test data for 93EDG-A, 93EDG-B and 93-EDG-D has been completed and it has been determined that the deficiency does not extend to those EDGs. All surveillance and testing requirements are current for 93EDG-A, 93EDG-B and 93EDG-D, therefore, the "B" train of emergency power will remain OPERABLE during the requested LCO extension, and 93EDG-A will remain available.

By granting the one-time allowance of 17 days for completion of TS 3.8.1 Required Action B.4, unnecessary challenges to plant operations personnel performing a plant shutdown will be avoided.

JAF AC Power Design

Offsite Power

During power operation, the JAF emergency power buses (10500 and 10600) are normally supplied by Normal Station Service Transformer (NSST) 71T-4 through separate feeder breakers, each of which supplies a non-vital bus (10300 and 10400) that feeds the associated emergency power bus (See Figure S71-002.cdr in Attachment 4). Should the plant trip for any reason, the feeder breakers from 71T-4 are tripped and feeder breakers from Reserve Station Service Transformers (RSST) 71T-2 and 71T-3 are closed such that each emergency bus is supplied by a separate RSST through the non-vital bus. The RSSTs are supplied by the 115 kV offsite power system. The 115 kV offsite power system is supplied by two (2) independent lines. One line (Line 4) receives power from the Oswego substation via the Nine Mile Point switchyard and the second line (Line 3) is supplied directly from the Lighthouse Hill hydro-electric power station. These lines come into the JAF 115 kV switchyard and are connected through motor operated disconnect 10017. This normally closed disconnect allows either line to supply power to both RSSTs such that power would be available to both emergency buses in the event that one offsite power source is lost.

Emergency Power

The JAF emergency power system (Figure S93-002.cdr in Attachment 4) consists of four (4) EDGs each located in a separate room within the EDG building, connected to the "A" and "B" emergency busses to supply emergency power during a loss of offsite power. Each of the two (2) 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 ECCS load circuit breakers. The EDGs are designed to provide an alternate, onsite source of reliable 4160 VAC power for safe shutdown equipment required to mitigate the consequences of a design basis accident in the event of a total loss of the normal and offsite power sources. Each generator has a continuous rating of 2,600 kW; therefore, the total loading capacity available per division, with both EDGs in the divisional pair operating, is 5200 kW at 4160VAC and 60Hz. Each EDG also has short time rating of 2,850 kW for 2,000 hours, 2,950 kW for 160 hours and 3,050 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, operators have transferred normal 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 93EDG-A (refer to Compensatory Measures discussed below).

Each generator has sufficient capacity to supply the required loads necessary to achieve safe reactor shutdown during an operational transient [i.e., loss of offsite power (LOOP) or degraded 4160 VAC emergency bus voltage]. The JAF EDG train availability is maintained by automatically limiting the initial loading of the single EDG while maintaining all emergency loads available.

In conclusion, the JAF plant design provides multiple and diverse means of supplying both normal and emergency power to the 4160V buses.

Coping Strategies

Abnormal Operating Procedures (AOPs) address the loss of individual 4160 VAC buses, the loss of station batteries, and in the worst case, Station Blackout. 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, the plant also has a strategy of extending the station blackout coping time. Technical Support Guideline (TSG) TSG-8, "Extending Station Blackout Time", provides guidance on this strategy. 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 DC power available. All necessary equipment is pre-staged.

2.0 Assessment

The James A. FitzPatrick Nuclear Power Plant (JAF) Technical Specification (TS) 3.8.1, "AC Sources – Operating," requires two qualified circuits between the offsite transmission network and the onsite Class 1E AC electrical power distribution system and two emergency diesel generator (EDG) subsystems to be operable in Modes 1, 2, and 3.

At 1015 on May 26, 2009, the plant entered Condition B of LCO 3.8.1 to support planned maintenance and inspection activities on 93EDG-C. On the second night of the maintenance activities, during a preventive maintenance task to megger the EDG rotor, a low reading was obtained. The reading indicated a potential fault on the rotor. Subsequent inspection and testing has determined that one of eight poles on the rotor was faulted. The rotor has been removed and transported to an approved vendor facility where additional inspection and testing has determined that it is necessary to rewind the faulted pole on the rotor. Using industry standard repair methodologies, the faulted pole is presently being rewound and will subsequently be tested.

TS 3.8.1 Condition B, One EDG subsystem inoperable, Required Action B.4 states, "Restore EDG subsystem to Operable status" and the associated Completion Time is "14 days AND 21 days from discovery of failure to meet LCO". TS 3.8.1, Required Actions F.1 and F.2 require that, if the required actions and completion times of Condition B are not met, the plant be placed in Mode 3 within 12 hours and in Mode 4 within 36 hours.

The current estimate for return of the EDG generator rotor from the rewind vendor is Sunday, June 7th. The rotor will have to be received on-site, inspected and transported to the EDG building in preparation for re-installation. It is expected that re-installation activities will commence late on Sunday, June 7th, once the rotor is released for re-installation. The re-installation window for the rotor and reassembly of the generator is estimated to be approximately 24 hours. Remaining maintenance activities which were part of the original maintenance window but could not be performed with the generator disassembled are estimated to take an additional 12 hours. Post-maintenance testing is estimated to be approximately 24 hours with an estimated completion on Tuesday, June 9th. Although this time line appears to be within the 14 day completion time allowed by LCO 3.8.1 Condition B, it is only an estimate that is based upon currently available information and does not include any allowances for unforeseen circumstances either at the rotor rewind vendor or the site.

Since the removal and re-installation of an EDG rotor is a first-time evolution at FitzPatrick, it is critical that the maintenance is performed in a deliberate manner without perceived time pressure. The pre-job briefings for the reassembly will clearly identify the expectation to stop work in the event that unanticipated circumstances arise or additional time is required to complete the specific task.

Based on the above, an extension of 3 days to the current 14 day Allowed Out-of-service Time (AOT) is requested. The 3 day extension will allow ample time to avoid undue time pressure to complete this first time evolution and will provide a reasonable period of time to resolve any unanticipated circumstances that may arise.

The Bases for TS 3.8.1 Condition B states, "The 14 day Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period." While the JAF Probabilistic Risk Assessment (PRA) Model has not completed the Regulatory Guide 1.200 peer review process, the risk model was used to assess the proposed completion time and the delta Core Damage Frequency (CDF) is 1.25E-06/ry.

JAF's request to allow a one-time use of a 17 day completion time allows time to complete the required repairs without maneuvering the plant. During this period additional compensatory measures will be implemented to minimize risk to the plant. These measures are identified on pages 17 through 19 of this attachment.

PRA Quality

As noted above, the JAF PRA Model has not completed the Regulatory Guide 1.200 peer review process. However, the scope, level of detail, and quality of the JAF PRA are sufficient to support a technically defensible and realistic evaluation of the risk change for this proposed completion time extension. The JAF PRA addresses internal events at full power only.

The JAF PRA is based on the original JAF PRA that was performed to support the Individual Plant Examination (1991). Since 1991, several updates have been made to incorporate plant design and procedure changes, update plant-specific reliability and unavailability data, improve the fidelity of the model, and incorporate Boiling Water Reactor Owners' Group (BWROG) peer review comments. Additional updates were made to the JAF PRA to support other applications, such as on-line maintenance, Integrated Leak Rate Test (ILRT) extension, risk-informed in-service inspection, and License Renewal.

The JAF PRA is maintained through a periodic review and update process. Peer certification of the JAF PRA using the BWROG peer review certification guidelines was performed in December 1997. Certification was performed by a team of independent PRA experts from U.S. nuclear utility PRA groups and PRA consulting organizations. This intensive peer review involved approximately two person-months of engineering effort by the review team and provided a comprehensive assessment of the strengths and limitations of each element of the PRA. On the basis of its evaluation, the certification team determined that, with certain findings and observations addressed, the quality of all elements of the PRA would be of sufficient quality to support risk significant evaluations with defense-in-depth input.

Facts and Observation sheets documented the peer review teams' insights and potential level of significance. All issues and observations from the BWROG Peer Review (i.e., Level A, B, C, and D observations) have been addressed and incorporated into the PRA model used for the JAF License Renewal project SAMA (Severe Accident Mitigating Alternatives) analysis (JAF PRA Model Revision 2, October 2004). The current PRA model (JAF PRA Model Revision 3, May 2007) was updated to include the plant design and procedural changes and component failure data of the Mitigating System Performance Indicators (MSPI) systems and was used for June 2007 NRC CDBI inspection.

To meet the requirement of the Regulatory Guide 1.200, the latest updated JAF PRA model (JAF PRA Model Revision 4) was developed. The major model changes that were incorporated into the JAF PRA since the last version can be summarized as follows:

- Updated the PRA model to reflect plant modifications and procedural changes.
- Updated the initiating event frequencies and component failure data by using generic data in NUREG/CR-6928, "Industry Average Performance for Components and Initiating Events at U. S. Commercial Nuclear Power Plants", February 2007 and performed the Bayesian update with the plant data. Updated CCF data by using generic data taken from U.S. Nuclear Regulatory Commission, "CCF Parameter Estimations, 2007 Update."
- Updated the offsite power recovery model 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, which contains data through

2004. In addition, the JAF PRA Update was supplemented with EPRI loss of offsite power events data from 2005 to December 2007.

- Revised the core damage definition from the peak clad temperatures greater than or equal to 2200°F to 1800°F defined in American Society of Mechanical Engineers, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME RA-Sb-2005, December 30, 2005.

- Updated the Human Reliability Analysis methodology from THERP (NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," October 1983) to EPRI HRA Method (EPRI-TR-100259, An Approach to the Analysis of Operator Actions in Probabilistic Risk Assessment,").

- Updated the internal flooding frequencies using pipe failure data analyses provided in EPRI report TR-1013141, "Pipe Rupture Frequency for Internal Flooding PRAs, Revision 1", March 2006.

- The accident sequence quantification truncation limit has been lowered from 10^{-11} to 10^{-12}.

- Enhanced the PRA model to incorporate the insights from the Vermont Yankee and Pilgrim Regulatory Guide 1.200 peer reviews, performed by the BWROG.

This updated PRA model has undergone an internal PRA group peer check and is currently scheduled for a Regulatory Guide 1.200 peer review, by the BWROG, during September 2009.

PRA Self-Assessment on ASME Standard Requirements

A 'gap assessment' (self assessment) was performed in June 2009 on revision 4 of the JAF PRA Model against the requirements of the ASME PRA standard for internal events (American Society of Mechanical Engineers, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME RA-Sb-2005, December 30, 2005). This self-assessment conforms to the suggested guidance found in RG 1.200. The self-assessment consisted of a two-stage process. The first stage involved addressing all of the VY and Pilgrim Station supporting requirements (SRs) designated as not meeting Capability Category II of the ASME PRA standard for internal events (SR not met). The second stage involved an assessment of the updated JAF PRA model against the ASME PRA standard to identify any 'gaps' that remained after addressing the VY and Pilgrim RG 1.200 Peer Review team findings.

The self assessment identified gaps in the following Supporting Requirements:

- Initiating Events Analysis (IE)

- Accident Sequence Analysis (AS)
- Success Criteria (SC)
- Systems Analysis (SY)
- Human Reliability Analysis (HR)
- Data Analysis (DA)
- Internal Flooding (IF)
- Quantification (QU)
- Large Early Release Frequency (LERF) Analysis (LE)

The impact of these gaps on the proposed 93EDG-C extended AOT is provided in the enclosed table.

ASME Supporting Requirement (SR)	Capability Category II	Impact on Current Application
IE-D3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the initiating events analysis.	Following the guidance presented in EPRI-1016737 FitzPatrick is currently examining the developed generic list of model and plant specific candidates for model uncertainty. The unavailability of 93EDG-C is bounded within the model uncertainties for those items which may be impacted by 93EDG-C unavailability. These items include the following: • Grid stability (generic issue#1) • Operation of equipment after battery depletion (generic issue#2) • Impact of containment venting on core cooling system NPSH (generic issue#7)
AS-C3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the accident sequence analysis.	See Response to ASME Supporting Requirement (SR) IE-D3.

ASME Supporting Requirement (SR)	Capability Category II	Impact on Current Application
SC-C3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the success criteria analysis.	See Response to ASME Supporting Requirement (SR) IE-D3.
SY-C3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the systems analysis.	See Response to ASME Supporting Requirement (SR) IE-D3.
HR-I3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the human reliability analysis.	See Response to ASME Supporting Requirement (SR) IE-D3.
DA-E3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the data analysis.	See Response to ASME Supporting Requirement (SR) IE-D3.
IF-F3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flooding analysis	See Response to ASME Supporting Requirement (SR) IE-D3.
QU-F4	DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the quantification analysis	See Response to ASME Supporting Requirement (SR) IE-D3.

ASME Supporting Requirement (SR)	Capability Category II	Impact on Current Application
LE-C8b	REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions during accident progression that could reduce LERF. USE conservative or a combination of conservative and realistic treatment for non-significant accident progression sequences.	Extended 93EDG-C AOT has no impact on this supporting requirement. The EDGs are expected to operate in a mild environment during a core damage progression. Hence, no additional 'engineering analysis' are required to justify continued EDGs operation or operator actions related to ac/dc power restoration.
LE-C9b	REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions after containment failure that could reduce LERF. USE conservative or a combination of conservative and realistic treatment for non-significant accident progression sequences.	Extended 93EDG-C AOT has no impact on this supporting requirement. From a harsh environment aspect, a postulated containment failure impacts only plant equipment located inside the Reactor Building due to a harsh environment; also containment failure impacts those systems which take suction from the torus pool. Since the EDGs are located in a separate structure outside of the Reactor Building, a postulated containment failure cannot impact continued EDG operation.

ASME Supporting Requirement (SR)	Capability Category II	Impact on Current Application
LE-E1	SELECT parameter values for equipment and operator response in the accident progression analysis consistent with the applicable requirements of paragraphs 4.5.5 and 4.5.6 including consideration of the severe accident plant conditions, as appropriate for the level of detail of the analysis.	The unavailability of a single EDG impact is small when comparing the delta increase in CDF and LERF. Hence, the impact of a single EDG on selected severe accident parameter (for example, the impact on LERF due to an inability to depressurize the RPV) is expected to also be minimal because the remaining diesel in the 'A' train remains available along with the full capability of the 'B' train ac power supply. In addition, when these analyses are performed the selected parameter function is completely disabled. For the example mentioned above, RPV depressurization is not allowed to occur. This implies the unavailability of all four EDGs.
LE-E4	QUANTIFY LERF consistent with the applicable requirements of Tables 4.5.8-2(a), 4.5.8-2(b), and 4.5.8-2(c).	Extended 93EDG-C AOT has no impact on this supporting requirement. This supporting requirement examines the LERF quantification process (methodology).
LE-F1a	PERFORM a quantitative evaluation of the relative contribution to LERF from plant damage states and significant LERF contributors from Table 4.5.9-3.	Extended 93EDG-C AOT has no impact on this supporting requirement. This SR identifies those plant initiators important to the occurrence of a large early release. The unavailability of 93EDG-C is not applicable to this supporting requirement.

ASME Supporting Requirement (SR)	Capability Category II	Impact on Current Application
LE-F1b	REVIEW contributors for reasonableness (e.g., to assure excessive conservatisms have not skewed the results, level of plant specificity is appropriate for significant contributors, etc.).	Extended 93EDG-C AOT has no impact on this supporting requirement. The unavailability of a single EDG does not influence the relative plant contributor's identified in LE-F1a. In addition, this supporting requirement is also addressed by SR LE-F3 and LE-G4.
LE-F3	IDENTIFY contributors to LERF and characterize LERF uncertainties consistent with the applicable requirements of Tables 4.5.8-2(d) and 4.5.8-2(e).	See Response to ASME Supporting Requirement (SR) IE-D3.
LE-G4	DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the LERF analysis	See Response to ASME Supporting Requirement (SR) IE-D3.
LE-G6	DOCUMENT the quantitative definition used for significant accident progression sequence. If other than the definition used in Section 2, JUSTIFY the alternative.	Extended 93EDG-C AOT has no impact on this supporting requirement. This is strictly a documentation issue in defining what is meant by a significant accident progression sequence.

PRA Capability and Calculation of Risk Increase

Risk-informed support for the proposed change is based on an evaluation of PRA calculations performed to quantify the change in Core Damage Frequency (CDF), Large Early Release Frequency (LERF), Incremental Conditional Core Damage Probability (ICCDP) and Incremental Conditional Large Early Release Frequency (ICLERF) resulting from the extended AOT duration for the 93EDG-C.

PRA Analysis

In order to support the proposed change in the TS 3.8.1 requirements, a probabilistic risk assessment was performed given a 93EDG-C Allowable Outage Time (AOT) of 17 days. The risk assessment involved use of the average maintenance unavailability PRA model from the updated PRA model. The success criteria for the Emergency Diesel Generators is that sufficient power be supplied to each safeguard bus load during transients or Loss of Coolant Accidents (LOCAs). For all scenarios, one EDG can supply the required loads on its respective 4.16 kV bus to achieve and maintain shutdown conditions.

The risk assessment predicted no significant increase in risk during the period of 93EDG-C inoperability using the updated PRA model.

Loss of Offsite Power (LOOP) Frequency & Recovery

LOOP Initiating Event Frequency

The LOOP initiating event frequency was estimated using LOOP data from the time period of 1997 – 2007. Consistent with NUREG/CR-6890, the time period starting 1997 was used in estimating the LOOP frequencies since this time period shows no statistically significant trend in the overall LOOP. In addition, the information from NUREG/CR-6890 was supplemented with LOOP data from EPRI reports for 2005, 2006 and 2007. As a result, the updated LOOP frequency for JAF was estimated as 4.3E-2/yr.

In addition to loss of offsite power occurring as an initiating event, the JAF PRA Update (Rev. 4) models both random failure of offsite power following an initiating event as well as a consequential LOOP. The JAF PRA Update uses a consequential LOOP probability of 3.4E-3 for non-LOCA initiators and 2.2E-2 for LOCA initiators.

Off-site Power (OSP) Recovery

The updated OSP recovery analysis for JAF was performed using LOOP data from 1990 – 2007. The LOOP duration data from NUREG/CR-6890 (which contains data from 1986 – 2004) was supplemented with data from EPRI reports for 2005, 2006 and 2007. Consistent with this NUREG, losses of offsite power that did not result in a plant trip were excluded. The LOOP events were grouped into one of three categories: plant-centered, grid-related and weather-related. For the offsite power recovery analysis, the time of interest is the potential bus recovery time. For events in which the potential bus recovery time is not provided (e.g., EPRI reports), 25 minutes was added to the potential offsite power (switchyard) recovery time based on discussions with JAF Operations staff. The below table provides a list for resulting OSP non-recovery probabilities that were calculated:

Updated JAF Offsite Power Non-Recovery Probabilities (1990 - 2007)				
Time (Hours)	**Plant-Centered Non-Recovery Probability**	**Grid-Related Non-Recovery Probability**	**Weather-Related Non-Recovery Probability**	**Composite Non-Recovery Probability**
0	1.00E+00	1.00E+00	1.00E+00	1.00E+00
0.5	6.44E-01	8.60E-01	8.88E-01	8.14E-01
1	4.37E-01	6.61E-01	8.05E-01	6.21E-01
2	2.46E-01	4.03E-01	6.92E-01	3.86E-01
3	1.59E-01	2.63E-01	6.15E-01	2.63E-01
4	1.12E-01	1.81E-01	5.57E-01	1.90E-01
5	8.25E-02	1.31E-01	5.11E-01	1.45E-01
6	6.33E-02	9.72E-02	4.74E-01	1.14E-01
7	5.00E-02	7.42E-02	4.42E-01	9.29E-02
8	4.03E-02	5.79E-02	4.15E-01	7.73E-02
19	8.03E-03	8.19E-03	2.55E-01	2.43E-02
24	4.85E-03	4.35E-03	2.18E-01	1.84E-02

EDG Repair

It should be noted that the JAF PRA Update (Rev. 4) does not take credit for repair of a failed or otherwise unavailable EDG. Consistent with the updated model, no such credit was taken for EDG repair for this analysis either.

Internal Event Risk

The risk of continued JAF operation with 93EDG-C out-of-service beyond the current 14 day AOT as measured by the delta core damage frequency (CDF), incremental conditional core damage probability (ICCDP), delta large early release frequency (LERF) and incremental conditional large early release probability (ICLERP) for internal events is shown below:

	Updated Model
Delta CDF/ry	1.25E-06
ICCDP (17 days)	5.82E-08
Delta LERF	6.40E-08
ICLERP (17 days)	2.98E-09

These values are less than the ICCDP and ICLERP guidance thresholds of 5E-07 and 5E-08, respectively, identified in NRC RG 1.177 ("An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications", 1998).

Internal Flood

The JAF PSA Update (Rev. 4) includes internal flood scenarios. Therefore, the risk impact associated with having 93EDG-C out of service is accounted for in the estimated increase in internal events CDF, LERF, ICCDP and ICLERP.

External Event Risk

The Individual Plant Examination of External Events (IPEEE) was performed as a one-time assessment of the impact of external events and is not periodically updated. The evaluation can be used to quantify changes due to specific pieces of equipment being removed; however, full update to incorporate changes in methodology and plant modification would be manpower intensive. Therefore, this analysis uses the original IPEEE model and notes several changes that if fully incorporated would result in even greater margins than those quantified below.

Fire

A fire risk assessment of the 93EDG-C out of service configuration was performed using the JAF IPEEE fire PRA model. The JAF fire analysis was performed using EPRI's Fire PRA Implementation Guide. The IPEEE was initially submitted in June 1996. Additional modifications were made as a result of comment resolution in the methodology between EPRI and NRC in 1999. The NRC issued a SER in 2000.

The three major fire zones which contributed to approximately 60 percent of fire risk are the Cable Spreading Room (zone CS-1), Control Room (zone CR-1), and the Relay Rooms (zone RR-1). Thirty-seven fire zone cutsets contributing to each fire zone were reviewed. The conditional core damage probability (CCDP) for each fire zone with 93EDG-C out of service was calculated and populated to a spread sheet to calculate the fire zone core damage frequency. The evaluation shows that the increased core damage frequency due to 93EDG-C out of service will have a small impact on zone CDF contributions to the overall fire risk. The delta CDF is 9.36.E-07/ry and ICCDP for fire events is 4.36E-08 for 17 days.

Fire zones CT-2 (East Cable Tunnel), EG-6 (Emergency Diesel Switchgear Room), and BR-4 (Train B Battery Charger Room), were the dominantly impacted zones by 93EDG-C out of service. Fire zone CT-2 contains cables and conduits for Train B safe shutdown equipment. Should a fire occur in zone CT-2, the plant is shutdown from the main control room. The necessary actions are proceduralized in AOP-28 (Operation During Plant Fires).

Fire zone EG-6 (Emergency Diesel Switchgear Room North), contains cables for the residual heat removal, core spray, service water, EDG room vent and cooling, emergency power distribution, and emergency diesel generator systems. Should a fire occur in zone EG-6, the plant is shutdown from the main control room. The necessary actions are proceduralized in AOP-28 (Operation during Plant Fires).

Fire zone BR-4 (Train B battery charger room), contains cable for the main steam, condensate, emergency power distribution and battery rooms vent and cooling systems. Should a fire occur in zone BR-4, the plant is shutdown from the Control Room. The necessary actions are proceduralized in AOP-28 (Operation during Plant Fires).

This evaluation does not consider the plant improvements identified by the IPEEE that have been implemented. These include the following:

1. The IPEEE recommended relocating heat detectors in the cable spreading room to severely limit contribution from transient fires. In lieu of the hardware modification, a change was made to administrative procedures proscribing unattended combustible material in the room. This change in procedure reduces the CDF contribution from transient fires in the Cable Spreading Room.

2. In the IPEEE analysis, spurious actuation or failure due to hot shorts and open circuits within cable jackets was included with a conservatively high probability of occurrence of 1.0. However, in the latest fire PRA methodology for NFPA-805 compliance [NUREG/CR-6850], this probability is addressed by assigning a probability of occurrence based on the configuration of the cabling and nature of the short circuit. Open circuits are no longer considered, therefore reducing the impact of the cable damage assessment. JAF uses thermoset cables which have a high damage temperature.

   A conservative estimate considering this new methodology for worst-case failure mode probabilities of hot short circuits for thermoset cables in trays with control power transformer (typical of MCC circuits) results in a probability of failure of 0.05. This change would reduce the CDF contribution from transient fires in Cable Spreading Room and Reactor Building.

3. In the IPEEE analysis, the dominant scenario in the Control Room analysis is a generic control room fire with a forced evacuation and failure to properly shut down the plant by implementing abnormal operating procedures. The ignition frequency used for the IPEEE was 1.07E-02 per year. However, with almost 10 years of additional accumulated industry experience, this frequency has been reduced to 2.5E-03 per year [NUREG/CR-6850]. This change would reduce the CDF contribution from fires in Control Room.

Furthermore, the compensatory measures discussed on pages 17 through 19 will be implemented to minimize the probability of fire occurrence during 93EDG-C out of service.

Based on the above evaluation, including the compensatory measures identified on pages 18 and 19, there is minimal impact on plant risk during the 93EDG-C out of service due to fire.

Seismic

The JAF plant has been designed to accommodate a safe-shutdown earthquake (SSE) with 0.15g peak ground acceleration (PGA). The seismic analysis performed in the IPEEE study is intended to act as a performance check on the design, estimating seismic capacity beyond the SSE.

The seismic analysis methodology implemented for JAF satisfied the NRC requirements for performing a seismic IPEEE as presented in Generic Letter 88-20, Supplement 4. Seismic events were evaluated using the Seismic Margins Analysis (SMA) method. The SMA methodology uses a deterministic approach to identify the weakest components in terms of High Confidence Low Probability of Failure (HCLPF) during peak ground acceleration. A seismic margin can be expressed in terms of the earthquake motion level that compromises plant safety, the seismic margin assessment determines whether there is high confidence that the plant can survive a given earthquake. No core damage frequency sequences were quantified as part of the IPEEE seismic risk analysis.

The seismic analysis is dominated by seismic initiating events that lead to station blackout; specifically, seismic-induced station blackout sequences controlled by seismic-induced block wall failures in the EDG Building.

For the proposed extended LCO 3.8.1 Required Action B.4 Completion Time seismic-induced failure of the block walls remains the limiting failure. Since the block wall failure is the limiting failure the inoperable status of 93EDG-C during this period would not result in any significant change to the existing core damage contribution from seismic events.

Other External Hazards

The JAF IPEEE submittal, in addition to the internal fires and seismic events, examined a number of other external hazards:

- High Winds and Tornadoes
- External Flooding
- Transportation and Nearby Facility Accidents
- Aircraft Hazard

- Severe Weather & Lightning

No risks to the plant occasioned by high winds and tornadoes, external floods, ice, and hazardous chemical, transportation and nearby facility incidents were identified that might lead to significant risk increase during the extended AOT of 93EDG-C.

Uncertainty or Sensitivity Issues

The PRA analysis of the AOT extension is relatively insensitive to uncertainties. The analysis did not credit equipment repair, so there are no uncertainties to be evaluated for that issue. JAF requires that important systems be protected during an AOT and it was confirmed that common cause is not an issue for the remaining EDGs. Therefore, issues related to uncertainties should have no effect on the PRA analysis. However, two sensitivity cases of key inputs were performed. Sensitivity case #1 doubled the LOOP frequency, and Sensitivity case # 2 doubled the common cause failure probability of other operable EDGs. The ICCDPs are presented in the following tables. These sensitivity cases concluded that the ICCDPs are less than the ICCDP guidance threshold of 5E-07, identified in NRC RG 1.177 ("An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications", 1998).

JAF PRA Model, Rev 4	AOT (days)	ICCDP
Sensitivity #1 [1]	17	1.17E-07
Sensitivity #2 [2]	17	6.13E-08

[1] 93EDG-C OOS with loss of offsite power frequency doubled.
[2] 93EDG-C OOS with EDG common-cause failure probabilities doubled.

Configuration Risk Management

Changes to plant configuration due to corrective and preventive maintenance will be controlled in accordance with procedure EN-WM-104, On-Line Risk Assessment. This Entergy fleet procedure complies with the requirement of 10CFR50.65 (a)(4), Regulatory Guide 1.182, and NUMARC 93-01 and requires that prior to performing maintenance activities, risk assessment shall be performed to assess and manage the increase in risk that may result from proposed maintenance activities.

Compensatory Measures

As discussed previously the scope of the repair is limited to a single pole on the rotor. This has been confirmed by inspection and testing at an approved vendor facility. The requested one-time allowance of a 17 day completion time for LCO 3.8.1 Required Action B.4 provides adequate time to complete the rewind activity, reassemble the rotor, test the rotor, transport the rotor to JAF, re-install the rotor, and perform the required post-maintenance testing to restore the EDG to OPERABLE status. During the period of the extended out-of-service time, the “B” train of emergency power will remain OPERABLE, both qualified offsite circuits will be available, and the second EDG (93EDG-A) on the “A” train of emergency power will remain available. This configuration is discussed in the current design basis for the plant and is allowed for limited periods of time (LCO 3.8.1 Condition B). The Operations department trains on various scenarios relating to loss of power and off-normal plant conditions. The operations staff is familiar with this configuration and the limitations on an emergency bus with only one EDG available.

To ensure the health and safety of the public, the following risk management actions will be implemented to increase operator awareness of critical equipment to provide reasonable assurance that the assumptions in the risk model are maintained, and to minimize the likelihood of a transient for the duration of the proposed LCO period.

1. 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 work activities on protected equipment; and 3) Operations Manager approval for any emergent work involving protected equipment.

    - Emergency Diesel Generators 93EDG-A, 93EDG-B and 93EDG-D
    - Emergency Service Water Pumps 46P-2A and 46P-2B
    - 4160V Normal and Emergency Switchgear Buses 10300, 10400, 10500 and 10600
    - Station Batteries 71SB-1 and 71SB-2
    - Station Battery Chargers 71BC-1 and 71BC-2
    - 125-Vdc Control boards 71BCB-2A and 71BCB-2B
    - Main Transformers 71T-1A, and 71T-1B
    - Normal Station Service Transformer 71T-4
    - Reserve Station Service Transformers 71T-2, and 71T-3
    - North and South 115 kV Bus Reserve Station Service Transformer Disconnect Switches 71EDSC-10015, 71EDSC-10017, and 71EDSC-10025
    - RHR/RHRSW Loops “A” & “B”
    - HPCI pump 23P-1
    - RCIC pump 13P-1
    - Torus vent valves 27AOV-117 and 27AOV-118
    - Diesel Driven Fire Pump 76P-1
    - Diesel Driven Fire Pump 76P-4

2. 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.

3. 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.

4. 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)).

5. 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.

6. Transient combustible loading in these areas will be reviewed and any unnecessary transient combustibles will be removed.

7. 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.

8. Maintenance and surveillance activities which could lead to Main Turbine trip will be avoided.

9. The plant Operations crew and Maintenance staff will be briefed on these risk management measures.

10. As an enhancement to the existing communications protocols daily communications will take place between JAF Operations and the Grid Operator.

11. 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.

12. Operations will monitor weather conditions to assess potential impacts on plant conditions due to adverse weather conditions.

13. These compensatory measures will be promulgated to the operating crews in an operations department standing order.

Conclusion

Based upon this review, there is no significant increase in the incremental core damage probability or large early release probability during the proposed TS amendment extended LCO period while operating at power.

3.0 Regulatory Analysis

3.1 No Significant Hazards Consideration Determination

Entergy has performed a "no significant hazards consideration determination" for the proposed amendment focusing on the three standard considerations as set forth in 10 CFR 50.92(c), "Issuance of Amendment," as described 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". 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 JAF 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.

3.2 Applicable Regulatory Requirements / Criteria

While JAF was not built or licensed to 10 CFR 50 Appendix A, General Design Criteria (GDC), it was evaluated and determined to meet the intent of Appendix A. GDC-17 requires two independent power sources; the proposed amendment does not alter JAF's compliance with the intent of that criterion. The one-time allowance of a 17 day completion time for TS 3.8.1 Required Action B.4 does not change the requirement to restore the inoperable EDG to operable status.

In conclusion, based upon the considerations described above:

1. there is reasonable assurance that the health and safety of the public will not be adversely affected 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 detrimental to the common defense and security or to the health and safety of the public.

4.0 Environmental Evaluation

In accordance with 10 CFR 51.30 an environmental assessment for proposed actions, other than those for a standard design certification under 10 CFR 52 or a manufacturing license under Part 52, shall identify the proposed action and include:

1. A brief discussion of:
   i. The need for the proposed action;
   ii. Alternatives as required by section 102(2)(E) of NEPA;
   iii. The environmental impacts of the proposed action and alternatives as appropriate; and
2. A list of agencies and persons consulted, and identification of sources used.

Need for Proposed Action

As previously stated in this submittal, a one-time amendment to the Technical Specifications will provide sufficient time to repair and test the "C" EDG. By granting a one-time allowance, the increase in transient risk encountered during a plant shutdown with only one emergency diesel generator subsystem available as required by TS 3.8.1.F will be avoided.

Alternatives Required by Section 102(2)(E) of NEPA

No alternatives are required by section 102(2)(E) of NEPA for this action.

Environmental Impacts of Proposed Action

Environmental Effluents

There is no change in the types of effluents or increase in the amounts of effluents, radioactive or non-radioactive, that are being, or may be released to the environment. The proposed TS amendment does not affect the generation of any effluent, nor does it affect any of the permitted release paths.

Radiation Exposure

There is no increase in individual or cumulative, occupational or public radiation exposure or planned increase in radiation exposure as a result of the planned EDG repairs during the proposed TS amendment extended LCO period. The EDG subsystem and the associated maintenance activities do not affect plant radiation levels, and therefore, do not affect dose rates and occupational exposure.

Risk of Radioactive Release

Although the JAF PRA Model has not been evaluated through the Regulatory Guide 1.200 peer review process at this time, it was used to evaluate the requested one-time allowance of a 17 day completion time for TS 3.8.1 Required Action B.4 from a probabilistic risk standpoint. This assessment considered the expected plant configuration during the period of the extended LCO and determined that it does not involve a significant increase in risk. The risk of continued JAF operation with the "C" EDG out of service during the additional 3 day period beyond the Technical Specification 14-day Completion Time, as measured by the Incremental Core Damage Probability (ICCDP), is 5.82E-08, and an ICLERP of 2.98E-09 which is below the ICLERP guidance of 5E-08 for internal events. This value is below the ICCDP guidance of 5E-07 and 5E-08 for ICLERP identified in NRC Regulatory Guide 1.177, "An Approach for Plant Specific, Risk Informed Decision making: Technical Specifications", 1998. The ICCDP for seismic, fire and flood external events is bounded by the ICCDP for internal events, and therefore, also meets the guidance threshold. Based upon this review, there is no significant increase in the incremental core damage probability or large early release probability during the proposed TS amendment extended LCO period while operating at power.

Therefore, Entergy has concluded that the proposed action will not involve additional direct, indirect, or cumulative impact to the environment, cultural, or historic resources, threatened or endangered species, or critical habitat. No environmental resources are affected by the proposed TS Amendment.

Agencies and Personnel Contacted

No Federal or State agencies were consulted during the preparation of this environmental assessment based upon a finding of no impact.

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Attachment 2

Proposed Technical Specification Changes (Mark up)

Pages

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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) <u>Maximum Power Level</u>

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

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

The Technical Specifications contained in Appendix A , as revised through Amendment No. ~~293~~ 294, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) <u>Fire Protection</u>

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.

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Attachment 3

Proposed Technical Specification Changes
(Final Typed)

<u>Pages</u>
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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:

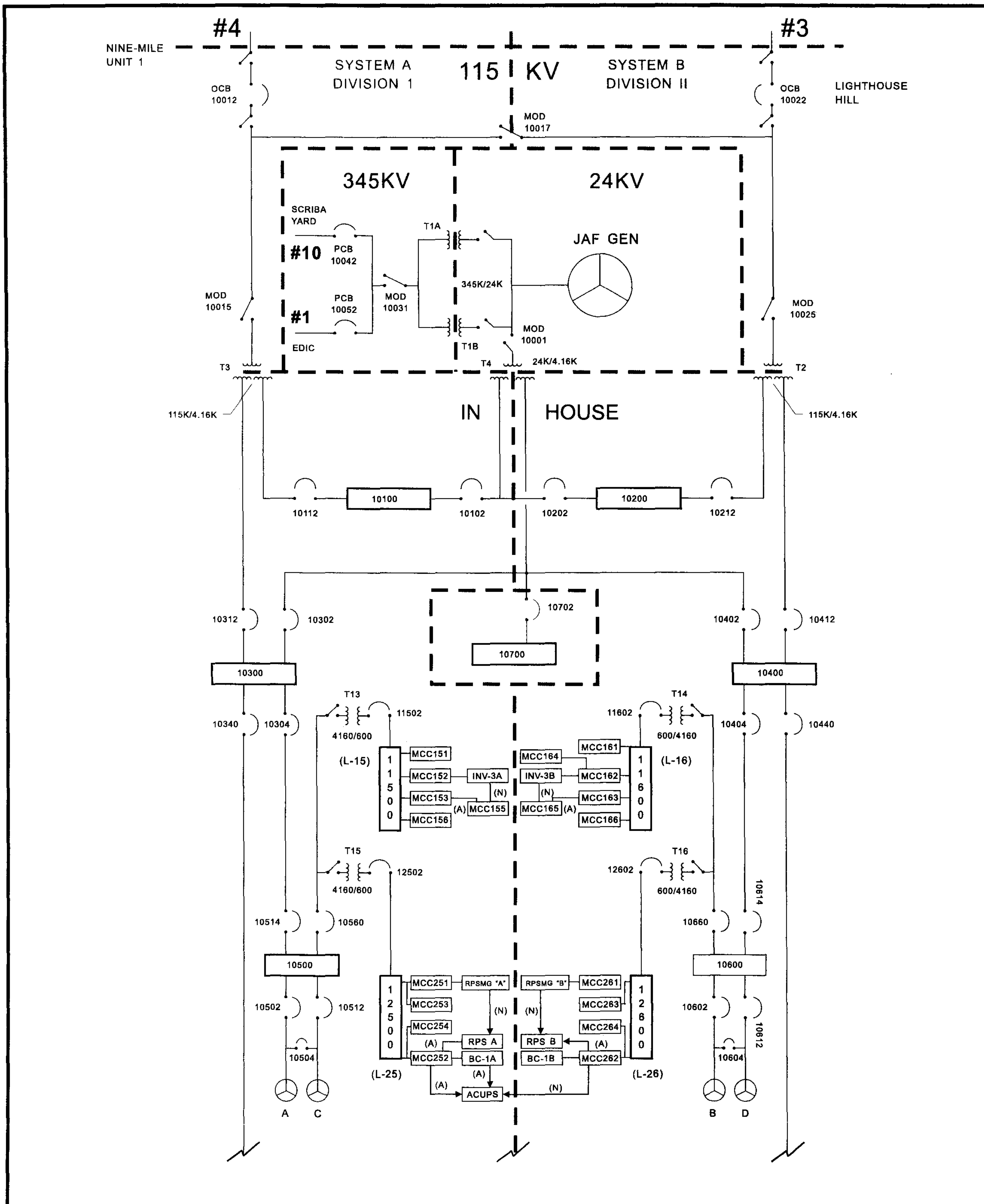
ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 Restore EDG subsystem to OPERABLE status.	14 days[1] AND 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. AND 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. AND 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. OR	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.

JAFP-09-0070
Attachment 4

Simplified Electrical Distribution Diagrams



Entergy

James A. FitzPatrick Nuclear Power Plant

FOR INFORMATION ONLY.

PROJECT

AC Distribution

DWG #

S71-002.cdr

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PROJ: **Electrical Distribution Emergency Diesel Generators**

DWG # **S93-002.cdr**