June 13, 2003

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing & Regulatory Programs
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT REGARDING TECHNICAL SPECIFICATION CHANGE REQUEST FOR EMERGENCY DIESEL GENERATOR ALLOWED OUTAGE TIME EXTENSION (TAC NO. MB5616)

Dear Mr. Young:

The Commission has issued the enclosed Amendment No. 207 to Facility Operating License No. DPR-72 for Crystal River Unit 3 (CR-3). The amendment consists of changes to the existing Technical Specifications in response to your letter dated July 3, 2002, as supplemented September 24, 2002, January 10, 2003, and March 20, 2003.

The amendment revises the CR-3 Improved Technical Specification (ITS) 3.8.1 and associated Bases, "AC Sources - Operating," by extending the allowed outage time for the emergency diesel generators (EDGs) from 72 hours to 14 days and modifies a note for two EDG ITS Surveillance Requirements.

A copy of the Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Brenda L. Mozafari, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:
1. Amendment No. 207 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

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CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEMINOLE ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 207
License No. DPR-72

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

   A. The application for amendment by Florida Power Corporation, et al. (the licensees), dated July 3, 2002, as supplemented September 24, 2002, January 10, 2003, and March 20, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-72 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 207, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance except for installation of an Aac source. An Aac source as described in the licensee's application supplement dated March 20, 2003, shall be installed before completion of refueling outage 14, as discussed in the NRC Safety Evaluation dated June 13, 2003. Implementation shall include incorporation of a description of the Aac source into the next scheduled Final Safety Analysis Report update submitted to the NRC pursuant to 10 CFR 50.71 (e) after the Aac installation.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Allen G. Howe, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 13, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 207

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of the Appendix "A" Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. Bases pages are included for information only.

Remove	Insert
3.8-2	3.8-2
3.8-3	3.8-3
3.8-4	3.8-4
3.8-8	3.8-8
3.8-10	3.8-10
B 3.8-7	B 3.8-7
B 3.8-8	B 3.8-8
B 3.8-10	B 3.8-10
---	B 3.8-10A
---	B 3.8-10B
---	B 3.8-10C
B 3.8-20	B 3.8-20
B 3.8-21	B 3.8-21
B 3.8-22	B 3.8-22
B 3.8-23	B 3.8-23

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

On July 3, 2002, Florida Power Corporation (FPC, the licensee, also doing business as Progress Energy Florida, Inc.) submitted a request for an amendment to Facility Operating License No. DPR-72 for Crystal River Unit 3 (CR-3). Specifically, the licensee requested a modification of the CR-3 Improved Technical Specifications (ITS) related to the emergency diesel generator (EDG) limiting conditions for operation (LCOs) action statements to extend the allowed outage time (AOT) for the EDG from 72 hours to 14 days and to modify two EDG Surveillance Requirements (SRs) to allow performance of the SRs at power if the SRs are required to demonstrate EDG operability. The AOT change would allow scheduled EDG maintenance to be performed while the unit is operating at power. This change would support both corrective and preventative maintenance.

The U.S. Nuclear Regulatory Commission (NRC) staff requested the licensee to provide additional information in order to complete its review. By letters dated September 24, 2002, January 10, 2003, and March 20, 2003, the licensee provided supplemental information that contained clarifying information only and did not change the initial no significant hazards consideration determination or expand the scope of the initial *Federal Register* notice.

The licensee states the following reasons for requesting the amendment: Many plants, including CR-3, limit planned equipment unavailability to approximately half of the applicable AOT. Thus, a 72-hour AOT provides a work and test window of approximately 35 to 40 hours. The 72-hour completion time is often very limiting in terms of corrective maintenance that can be accomplished without a plant shutdown. CR-3 has experienced several situations where corrective maintenance has challenged the 72-hour window. On these occasions, CR-3 has begun the process of requesting enforcement discretion to avoid an unnecessary shutdown. The proposed 14-day AOT provides a 7-day work window, which is long enough to accomplish all planned preventative maintenance activities.

The licensee states that a longer AOT will reduce the probability that enforcement discretion will be needed in the future and minimize the potential for requiring relief where there would not be an opportunity for public comment. According to the licensee, therefore, this change would reduce the potential administrative burden associated with the enforcement discretion process for both the utility and the NRC.

The proposed AOT extension is based on the findings of both deterministic and probabilistic risk assessment. The NRC staff has reviewed the proposed changes and finds them acceptable for the reasons discussed in the following evaluation.

2.0 REGULATORY EVALUATION

The regulatory requirements that the NRC staff applied in its review of the amendment included General Design Criterion (GDC) 17, "Electric Power System," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," which requires that an onsite electric power system and an offsite electric power system be provided to permit functioning of structures, systems, and components important to safety. The CR-3 electrical distribution system has been previously evaluated to conform to Criterion 39, "Emergency Power for Engineered Safety Features," proposed by the Atomic Energy Commission in a rulemaking for 10 CFR 50 published in the *Federal Register* on July 11, 1967. The licensee states in Section 8.1 of the Final Safety Analysis Report (FSAR) that the electrical systems for Crystal River Unit 3 are in compliance with the intent of 10 CFR Part 50, Appendix A, General Design Criterion 17. Therefore, 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.

According to 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," preventive maintenance activities must not reduce the overall availability of the systems, structures, and components.

Pursuant to 10 CFR 50.36, "Technical Specification," a licensee's TS are required to establish LCO action statements that include AOTs for equipment that is required for safe operation of the facility.

Regulatory Guide (RG) 1.93, "Availability of Electric Power Sources," provides guidance with respect to operating restrictions (i.e., AOTs) if the number of available alternating current (ac) sources is less than that required by the TS LCO. In particular, the guidance recommends a maximum AOT of 72 hours for an inoperable ac source.

As described by the licensee's July 3, 2002, application, CR-3 is a single nuclear unit located on a site with four fossil units. Units 1 and 2 are immediately adjacent to and supply auxiliary steam, water purification, and other support services to CR-3. Units 4 and 5 are approximately 1000 yards north of CR-3. Units 1, 2, and 4 deliver power to a shared 230 kV switchyard located between the north and south plants. Units 3 and 5 deliver power to a shared 500 kV switchyard located east of the 230 kV yard. Seven transmission lines carry power to and from the site and intertie with the state grid at four different points. Five of these offsite transmission lines (Newberry, Central Florida, Holder, CR East and Brookridge) carry power to and from the 230 kV switchyard. Two 500 kV lines (Brookridge and Central Florida) carry power to and from the 500 kV switchyard. CR-3 receives offsite power from the 230 kV switchyard through two independent, dedicated transformers, the offsite power transformer (OPT) and the backup engineered safeguards transformer (BEST), each capable of supplying either or both trains of engineered safeguards (ES) buses. The normal operating alignment is that the BEST is aligned to the "B" ES train and the OPT is aligned to the "A" ES train. Each individual circuit feeder breaker (4900 and 4902 for the OPT, 1691 and 1692 for the BEST) is capable of handling the required transformer loading.

The CR-3 main generator delivers power to the 500 kV switchyard. This design feature provides independence between electrical output and its source of offsite ac power (230 kV switchyard) for CR-3. Therefore, CR-3 is not susceptible to a loss of offsite power resulting from a grid disturbance due to a reactor trip. The loss of generation at CR-3 does not impact the voltage of the offsite power supplies' non-safety or ES busses at CR-3. Although CR-3 cannot be powered from the 500kV switchyard during power operation, a manual connection to this switchyard can be made with the generator offline. This process removes links between the switchyard and the main generator and can connect all station busses (within loading limits) to the 500kV switchyard by backfeeding through the step-up and unit auxiliary transformers. Establishing back-end operation takes approximately 8 hours.

The onsite standby power source for each ES bus is a dedicated EDG that starts automatically on a loss-of-coolant accident (LOCA) signal or an engineered safety feature bus degraded voltage or under voltage signal. The EDGs are sized so that either one can carry the required ES load. Each EDG unit will feed its designated ES bus. The onsite ac emergency power system has the required redundancy; meets the single failure criterion; is testable; and has the capacity, capability, and reliability to supply power to all required safety loads.

In addition, CR-3 will add alternate ac source (Aac) capability. The extended EDG AOT will only be utilized when the Aac source is available. The Aac source can be manually aligned to supply the applicable safety-related bus with simple operator action.

3.0 TECHNICAL EVALUATION

The proposed ITS changes are as follow:

(1) The Completion Time for ITS 3.8.1, Condition A, one required offsite circuit inoperable, Required Action A.3, "Restore required offsite circuit to OPERABLE status" is to be revised from:

"72 hours AND 6 days from discovery of failure to meet LCO" to

"72 hours AND 6 days from discovery of failure to meet LCO OR 17 days if alternate AC power is available"

(2) The Completion Time for ITS 3.8.1, Condition B, one EDG inoperable, Required Action B.4, "Restore EDG to OPERABLE status" is to be revised from:

"72 hours AND 6 days from discovery of failure to meet LCO" to

"72 hours AND 6 days from discovery of failure to meet LCO OR 14 days if alternate AC power is available* AND 17 days from discovery of failure to meet LCO"

* On a one-time basis, each EDG may be inoperable for up to 14 days without alternate AC available. The ability to apply the one-time 14-day Completion Time to each EDG will expire on May 15, 2004.

(3) Changes to SR 3.8.1.8, Note 1 and SR 3.8.1.11, Note 2 would result in the following:

This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

The proposed changes are based on the findings of both deterministic and probabilistic safety assessment (PSA). Although the licensee has demonstrated, based on its PSA , that the proposed changes would result in a slight increase in core damage frequency, incremental conditional core damage probability, large early release frequency, and incremental conditional large early release probability. The licensee states that these increases are well below values that are considered risk significant in accordance with current RG guidance (see Section 4.0 for PSA Evaluation). While the NRC staff has reviewed the proposed amendment from a probabilistic viewpoint as set forth in Section 4 of this SE, it has also reviewed it from a deterministic approach as follows:

3.1 <u>EDG Reliability and Unavailability</u>

The NRC staff evaluated the proposed change to ensure that the overall availability of the EDG will not be significantly reduced as a result of increased online preventive maintenance activities, and that the 14-day AOT will be consistent with the objective of the Maintenance Rule in 10 CFR 50.65. In response to the NRC staff's concern regarding the reliability and availability of the EDGs and the impact of the AOT extension on EDG unavailability per the Maintenance Rule, the licensee stated that a specific procedure, PT-354, "EDG Reliability and Unavailability Program," was established to ensure compliance with the station blackout (SBO) coping analysis and addressed this issue. The licensee indicated that there have been no failures in the past 20 demands, one failure in the past 50 demands, and three in the past 100 demands. The period covered by the 100 start demand window includes March 18, 1999, through September 30, 2002. The unavailability for the three-quarter period ending September 30, 2002, is 0.73 percent. The EDG Reliability Program records unavailability for a calender year; however, the World Association of Nuclear Operators and NRC Reactor Oversight Program (ROP) performance indicators (PIs) and the Maintenance Rule program record rolling multiple-year averages. The 3-year average unavailability reported under the ROP PI for the period ending September 30, 2002, is approximately 2.2 percent. This is roughly 0.7 percent higher than typical due to a 345-hour T/2 Fault Exposure impact reported in conjunction with a July 5, 2001, start failure event. In summary, there have been only three EDG failures in the past 4 years, and the 3-year average unavailability performance has met the goals established/committed, even with the T/2 Fault exposure. The above indicates that the EDGs at CR-3 are reliable, and the EDG unavailability is adequately controlled.

The licensee states that a small increase in unavailability as defined under the Maintenance Rule is expected due to the requested amendment. This projected increase is due to the fact that the offline inspections are not reportable under the Maintenance Rule, provided they are performed during a window when the train is not required (only one EDG is required while shut down in Modes 5 and 6). Should the AOT extension be granted, the work normally performed during outages would be relocated outside the outage windows. As a result, reported 2-year averaged Maintenance Rule unavailability would increase by approximately 1 percent. Because the unavailability of the EDG is monitored under the implementation of the Maintenance Rule, the NRC staff finds the above increase in unavailability to be reasonable.

3.2 Offsite Power System

In response to the NRC staff's concern regarding loss of offsite power (LOOP) events at CR-3, on January 10, 2003, the licensee stated that there were a number of events where offsite power was lost to one or more ES busses at CR-3. Except for the two recent events, all the LOOP events involving the 230 kV switchyard occurred prior to the addition of both the OPT and the BEST. Several LOOP events occurred in 1993 while the plant was shut down and aligned to the 500 kV switchyard. Back-end from the 500 kV switchyard is not available during power operation, although it is capable of powering either or both ES busses during shutdown. Events that occurred in this configuration cannot occur during unit operation because offsite power is provided only from the 230 kV switchyard during power operations. Only one event has occurred where offsite power could not have been restored to both ES busses within approximately 2 minutes. The March 27, 1992, event would have required approximately 20 minutes to restore power from the Units 1 and 2 startup transformer due to relay failures. The two recent events of partial LOOP were caused by damage to the OPT cables. In order to prevent recurrence of these failures, the cables have been replaced from the OPT to the west berm termination enclosure (approximately 1200 feet) and the lightning protection has been enhanced. Based on evaluation of existing performance data and enhancements to improve reliability of the system, the NRC staff finds that an acceptable offsite power system will be available during the extended EDG outage.

3.3 Station Blackout

The NRC staff allowed an EDG AOT of 14 days for plants that added an Aac source to meet the requirements of 10 CFR 50.63, "Loss of all alternating currents," and plants that provided a temporary ac power source during the extended AOT period. The NRC staff's review indicated that CR-3 is a 4-hour coping plant that does not have an Aac source. The NRC staff requested information on what method of Aac source would be made to ensure defense in depth was maintained during EDG extended maintenance. In response to the above concern, on March 20, 2003, the licensee stated that CR-3 will add an Aac source capability prior to completion of Refueling Outage 14, currently scheduled for fall 2005. The Aac source is intended to provide defense in depth during EDG online maintenance and other times when it is available and is not intended to be used to change the CR-3 licensing basis for compliance with SBO. Several options for providing Aac source are under consideration. Each of these options will meet the following criteria:

1. The Aac source will not be designed to meet Class 1E or safety system requirements.

2. The Aac source will not be protected against the effects of failure or misoperation of mechanical equipment, including fire, pipe whip, jet impingement, water spray, flooding from pipe break, radiation, pressurization, elevated temperature or humidity caused by high or medium energy pipe break, missiles, or seismic events.

3. Aac source components and subsystems shall be protected against the effects of likely weather-related events that may initiate a LOOP event. Permanent structures will be designed to meet the established building codes, which require the ability to withstand the effect of a Category 3 hurricane up to 110 mph winds and associated flooding. Temporary structures and equipment will be secured to withstand the severe weather events likely during the duration of their use.

4. The Aac source components will be physically separate from safety-related components.

5. Failure of the Aac source components will not adversely affect Class 1E power systems.

6. The Aac source shall have electrical separation from the Class 1E power system by two circuit breakers in series, one of which will be a 1E circuit breaker at the Class 1E bus.

7. The Aac source will not normally be connected to the onsite or offsite power systems. The Aac source will not be capable of automatic loading of shutdown equipment.

8. The Aac source will be designed to minimize potential for common cause failure including the following features:

    - The Aac source will have a direct current (dc) system separate from the Class 1E dc power system used to support the EDGs.

    - The Aac source will have an independent air start system that does not rely on offsite power or Class 1E power.

    - The Aac source will have a fuel supply separate from the Class 1E power system.

    - If the Aac source is identical to the emergency onsite power source, active failures of the emergency power source will be evaluated for applicability and corrective action.

    - No single point of vulnerability will exist whereby a likely weather event or single active failure could affect the onsite emergency ac source and the offsite sources and simultaneously cause failure of the Aac source.

    - The Aac source shall be capable of operating with a loss of all offsite and onsite ac power.

    - The Aac source will have appropriate post-maintenance testing following maintenance prior to relying on the Aac source for an extended AOT on the EDGs.

9. The Aac source will be capable of carrying the loads required for safe shutdown, including maintaining adequate voltage and frequency such that the performance of safety systems is not degraded.

10. The Aac source capacity and availability will be assured by plant procedures. Availability will be assured during an extended EDG AOT by the following:

- Starting the Aac source and assuring proper operation prior to removing the EDG from service,

- Verifying every 72 hours that a 24-hour fuel supply is onsite, and

- Ensuring the Aac source is electrically and mechanically ready for manual operation and can be aligned to supply the applicable safety-related bus with simple operator action every 72 hours.

11. An initial test will be performed to demonstrate that the Aac source can be aligned to the safety-related busses within 1 hour.

12. Extended EDG AOTs will only be utilized when the Aac source is available. The AOT reverts back to 72 hours if the Aac source is not available. If the Aac source becomes unavailable during an EDG extended AOT, the 72-hour AOT begins at the time of discovery that the Aac source was not available, not to exceed a total of 14 days from the time the EDG originally became inoperable.

The licensee is currently evaluating both permanent and temporary Aac source options. Any alternative selected will meet criteria 1 through 12 described above. The NRC staff finds that the licensee's commitment to have an Aac source meeting criteria 1 through 12 during an extended EDG outage is acceptable. The Aac source, as installed, will be described in the FSAR update following the Aac source installation.

3.4 <u>Other Considerations</u>

The NRC staff was concerned that EDG overhauls necessitate a number of tests that now will be conducted during power operation if this AOT extension is granted. In particular, many manufacturers recommend a full-load rejection test if a governor is replaced. Performing this test during power operation could cause voltage transients on the 4160 V safety bus. The licensee stated that the CR-3 ITS does not need a full-load rejection test as part of normal ITS SR or as part of the post-maintenance EDG testing. The licensee stated that following EDG governor maintenance, the single largest load rejection is performed and is simulated by establishing an EDG load of 800 kW from the grid, and then opening the output breaker. In this way, EDG response is kept from impacting the voltage and frequency of the safety-related bus. Further, the two most recent EDG governor replacements were April 1994 (EDG-1A, during R9) and April 1996 (EDG-1B, during R10). The review of the results of the single largest load rejection tests indicated that the voltage and frequency perturbation are minimum. Accordingly, the NRC staff finds that this concern is satisfactorily resolved.

The NRC staff requested information on how planning of the extended EDG maintenance should consider the time needed to complete the extended EDG maintenance and the ability to accurately forecast weather conditions that are expected to occur during the maintenance. In its response, the licensee stated that CR-3 will consider factors such as grid and weather conditions prior to entering an extended EDG maintenance outage. CR-3 does not have specific guidelines or contingencies developed for extended EDG maintenance. Engineering judgment would be used on a case-by-case basis as to what alternative would be most expedient to restore the EDG to service. Weather-related risks would be assessed prior to initiating extended maintenance and the timing of the maintenance window would be developed to ensure the maximum time possible is available for system restoration. Contingencies would be developed as appropriate at that time. Accordingly, the NRC staff finds that this concern is satisfactorily resolved.

In response to the NRC staff's concern whether the licensee's risk management procedures cover a comprehensive walkdown just prior to entering the period of reduced equipment availability (EDG extended maintenance on-line), the licensee stated that risk management procedures do not specifically address comprehensive walkdowns prior to reduced equipment availability.

The CR-3 Maintenance Rule program evaluates and manages risks due to maintenance. For high significance maintenance activities, compensatory measures are developed. For the EDG extended maintenance, CR-3 concurs that a thorough walkdown of redundant electrical and mechanical systems would be appropriate. The licensee commits to performing such a walkdown prior to extended EDG maintenance. Accordingly, the NRC staff finds that this concern is satisfactorily resolved.

3.5 Licensee Commitments

During the performance of extended maintenance on the EDGs, the licensee has committed in the letter dated March 20, 2003, to include the following additional precautions to minimize risk:

1. CR-3 will perform procedure CP-253, "Power Operation Risk Assessment and Management," which provides for both a deterministic and probabilistic evaluation of risk for the performance of all maintenance activities. This procedure uses the Level 1 PSA model to evaluate the impact of maintenance activities on core damage frequency. CR-3 will not plan any maintenance that results in "Higher Risk" (Orange Color Code) during EDG maintenance.

2. Emergency core cooling system (ECCS) equipment, emergency feedwater, control complex cooling and auxiliary feedwater (FWP-7 and MTDG-1) will be designated administratively as "protected" (no planned maintenance or discretionary equipment manipulation).

3. Prior to initiating a planned EDG extended outage, CR-3 will verify the availability of offsite power to the 230 kV switchyard and ensure that the capability to power both ES busses is available from each of the two ES offsite power transformers (OPT and BEST).

4. CR-3 will not initiate an EDG extended preventive maintenance outage if adverse weather, as designated by Emergency Preparedness procedures, is anticipated.

5. No elective maintenance will be scheduled in the switchyard that would challenge the availability of offsite power to the ES busses.

6. A periodic fire watch will be established in fire areas that are considered risk-significant by the individual plant examination of external events (IPEEE), affect both EDGs, or have increased risk significance due to EDG maintenance.

7. CR-3 will perform a comprehensive walkdown of redundant electrical and mechanical systems.

8. CR-3 will provide information on the method of Aac power that will be made available during EDG extended maintenance.

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

3.6 Other Surveillance Changes

In addition to the EDG AOT extension, FPC requests that Notes for two EDG SRs be modified to allow performance of the SR at power if the SR is required to demonstrate EDG OPERABILITY and an assessment determines that plant safety is maintained or enhanced. These changes are consistent with Technical Specification Task Force (TSTF) Traveler 283, Revision 3, and NUREG-1430, "Standard Technical Specifications - Babcock and Wilcox Plants, Revision 2." FPC has evaluated TSTF-283 and determined that the TSTF justification for the modified note applies to CR-3 SR 3.8.1.8 (SR 3.8.1.9 in the TSTF) and SR 3.8.1.11 (SR 3.8.1.14 in the TSTF). The modified note will provide the flexibility to perform the SRs following corrective maintenance (or other conditions described in the Bases for the SRs). This flexibility could prevent the need for a plant shutdown for certain EDG corrective maintenance activities such as a governor repair or replacement. To justify the performance of SR 3.8.1.8 (single largest load rejection test) online, the licensee performed dynamic analyses to simulate the largest load rejection cases. A database representative of a fully loaded EDG-1A, automatically connected plus manually applied essential loads, was created for this analysis. The database was run using dynamic analysis Industrial Power System Time Simulation Software. The voltage and frequency transient response of EDG-1A following an instantaneous load drop of 3256 kW to 2268 kW (an instantaneous load drop of 988 kW) was acceptable. The motor models and the EDG model used in the software have been field-validated by previous testing and are representative of the actual equipment in the field. A similar dynamic analysis was conducted for EDG-1B, and the voltage and frequency transient response following an instantaneous load drop of 3320 kW to 2344 kW (an instantaneous load drop of 976 kW) was found to be acceptable. The NRC staff finds that the voltage and frequency perturbation due to single largest load rejection test is minimal.

3.7 Deterministic Conclusion

The NRC staff evaluated the proposed change to extend the EDG AOT from 3 to 14 days. The NRC staff concludes that the licensee's request for the 14-day EDG AOT to perform major maintenance is acceptable. The NRC staff's conclusion is based on the following:

1. An Aac source will be available within 1 hour to support required safe-shutdown loads during the extended EDG AOT, consistent with 10 CFR 50.63. (Description of the Aac source in the FSAR is the subject of a condition on the implementation of the amendment.)

2. The extended AOT will be typically used to perform infrequent (i.e., once every 18 months) diesel manufacturer's recommended inspections and preventive maintenance activities.

3. The 14-day EDG AOT would reduce the entries into the LCO and reduce the number of EDG starts for major EDG maintenance activities.

4. The licensee will implement power operation risk assessment and management during the extended outage.

Further, the NRC staff believes that regulatory commitments to implement other restrictions and compensatory measures during the extended AOT would ensure the availability of the remaining sources of power and would minimize the occurrence of an SBO.

Also, the NRC staff finds the licensee's request for a one-time 14-day EDG AOT extension to be acceptable based on the following: (1) The extended AOT would reduce entries into the LCO and reduce the number of EDG starts for these major maintenance activities. (2) The licensee will take compensatory measures 1 through 7 as discussed in Section 3.5 of this SE during the extended AOT to minimize the occurrence of an SBO. (3) The licensee will implement power operation risk assessment and management during the extended outage.

Additionally, the NRC staff concludes that the proposed modification to SR 3.8.1.8, Note 1, and SR 3.8.1.11, Note 2, is acceptable, since the voltage and frequency transients caused by this surveillance do not have an adverse effect on safety-related loads. Furthermore, the required assessment (No. 4 above) will ensure no adverse effect on plant safety.

The NRC staff also concludes that the proposed changes will not affect the conformance of CR-3 with the intent of GDC 17 as described in Section 2 of this SE.

4.0 PROBABILISTIC SAFETY ASSESSMENT EVALUATION

The Probabilistic Safety Assessment Branch (SPSB) evaluated whether the proposed changes would have a significant risk impact. The following provides the PSA insights:

4.1 Tier 1 Analysis

An analysis was performed in a manner consistent with RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications." This analysis is for a risk-informed quantitative impact of the proposed permanent change in the EDG train from 72 hours to 14 days (336 hours). The licensee performed this evaluation as plant-specific using the CR-3 PSA model for online operation.

The licensee points out that the computed risks associated with EDG maintenance are largely coupled with the reliability of offsite power. The CR-3 PSA includes two LOOP initiators, 1E_T3 and 1E_T15. 1E_T3 is a loss of the 230 kV switchyard. This event results in a loss of all normal offsite power and a plant trip. 1E_T15 is a LOOP to the Startup Transformer and the BEST. This event results in a plant trip and a LOOP to the aligned ES bus (normally "B" train). Following a 1E_T15 event, offsite power is still available to the "A" ES bus from the OPT. A loss of feed from the OPT alone does not cause a plant trip or a significant challenge to normal power production. According to the licensee, this assumption was validated by an event on June 17, 2002, when the feed from the OPT was lost. The plant remained online throughout the event and the restoration of the OPT feed. Therefore, loss of the OPT feed alone is not modeled as an initiating event in the licensee's PSA because it does not result in a significant plant transient.

The licensee's computed incremental conditional core damage probability, averaged for preventive and corrective maintenance, is 4.53E-07, within the RG 1.177 guideline of 5E-07. The licensee's computed incremental conditional large early release probability, averaged for preventive and corrective maintenance, is 2.3E-10, within the RG 1.177 guideline of 5E-08.

The licensee's computed delta core damage frequency (CDF) is 9E-08/r-yr, and the computed delta large early release frequency (LERF) is <1E-9/r-yr. These fall below Region III of Figures 3 and 4 of RG 1.174, and are thus considered very small. The baseline CDF is 6.8E-06/r-yr, and the baseline LERF is 3.6E-07/r-yr. All of the above risk measures/calculational results meet the guidelines of RGs 1.177 and 1.174 and are acceptable to the NRC staff.

4.2 Tier 2 Discussion

To avoid the emergence or persistence of risk-significant configurations during the performance of extended maintenance on the EDGs, the licensee will take additional precautions to minimize risk. These include the following:

(1) Performance of procedure CP-253, "Power Operation Risk Assessment and Management," which requires both a deterministic and probabilistic evaluation of risk for the performance of all maintenance activities. This procedure uses the Level 1 PSA model to evaluate the impact of maintenance activities on CDF. The licensee will not plan maintenance that results in higher risk (Orange Color Code) during extended (>72 hours) EDG maintenance.

(2) ECCS equipment, emergency feedwater, and control complex cooling and auxiliary feedwater will be designated administratively as protected (no planned maintenance or discretionary equipment manipulation).

(3) Prior to initiating a planned EDG outage, the licensee will verify the availability of offsite power to the 230 kV switchyard and ensure that the capability to power both ES busses is available from each of the two ES offsite power transformers (OPT and BEST).

(4) CR-3 will not initiate an EDG extended preventive maintenance outage if adverse weather, as designated by Emergency Preparedness procedures, is anticipated.

(5) The licensee will schedule no elective maintenance in the switchyard that would challenge the availability of offsite power to the ES busses.

(6) A periodic fire watch (every few hours) will be established by the licensee in fire areas that are considered risk significant in the IPEEE, affect both EDGs, or have increased risk significance due to EDG maintenance.

The NRC staff judges that the licensee's Tier 2 precautions are adequate.

4.3 PSA Quality

The licensee models used for this application were generated using updated Individual Plant Examination (IPE) models developed is response to Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," and associated supplements. The original development work was a level one Probabilistic Risk Assessment study completed in 1987 (Crystal River Unit 3 Probabilistic Risk Assessment, Florida Power Corporation, Science Applications Intl. Corporation, July 1987), which was submitted to the NRC staff and reviewed by the Argonne National Laboratory (NUREG/CR-5245). This study was subsequently updated for the GL 88-20 IPE submittal to include a level two containment analysis and an internal flooding analysis. The study was subjected to reviews by the relevant CR-3 system engineers, and the review of the event sequence analysis by the Nuclear Safety Supervisor at CR-3, a former senior reactor operator.

Revisions to the models have been made to maintain the models consistent with plant design changes and operational changes. These changes have been made by individuals knowledgeable in risk assessment techniques and methods, and reviewed by plant Engineering and Operations personnel familiar with the plant design and operation.

Current administrative controls include written procedures and review of all model changes, data updates, and risk assessments performed using PSA methods and models. Risk assessments are performed by a PSA engineer, reviewed by another PSA engineer, and approved by the PSA Supervisor or designee.

Since the submittal of the original PSA study in 1987, the PSA models have been maintained consistent with the current plant configuration such that they are considered "living" models that reasonably reflect the as-built, as-operated plant. The PSA models are updated for different reasons, including plant changes and modifications, procedure changes, accrual of new plant data, discovery of modeling errors, and advances in PSA technology. The update process ensures that the applicable changes are implemented and documented in a timely manner so that risk analyses performed in support of plant operations reflect the current plant configuration, operating philosophy, and transient and component failure history. The PSA maintenance model and update process is described in the licensee's administrative procedure ADM-NGGC-0004, "Updates to PSA Models." Model updates are performed at a frequency dependent on the estimated impact of the accumulated changes. Guidance to determine the need for a model update is provided in the procedure. Prior to startup from a refueling outage, known outstanding changes, including identified modeling errors and enhancements, are reviewed, and either model changes are implemented or the outstanding item is dispositioned to be deferred for a future model update.

The NRC staff judges that the PSA model quality is adequate for this application as is the timeliness of identifying errors and enhancements.

4.4 PSA Software

Computer programs that process PSA model inputs are verified and validated by the licensee in accordance with administrative procedure CSP-NGGC-2505, “Software Quality Assurance and Configuration Control of Business Computer Systems.” This procedure provides for software verification and validation to ensure that the software meets the software requirement specifications and functional requirements, and typically includes a comparison of results generated in a particular analysis to the results generated with previously approved software.

Validation requirements for each quality-related PSA computer program are documented in the Software Life Cycle document, which consists of a Software Verification/Validation Plan (SVVP) and Report (SVVR). These requirements include the method of validation, the frequency of validation, the documentation required, and the acceptance criteria. Actual validation benchmark problems can exercise more than one program, but a separate SVVR must be submitted for each program. Each SVVP and SVVR is reviewed, and then approved by the software owner, who is the PSA Supervisor. Software validation tests both the software and the hardware. Validation tests are also performed following any significant change in the hardware, operating system, or program, or if the validation period established in the SVVP procedure expires.

The NRC staff judges that the licensee’s SVVP plan is adequate for this application.

4.5 Model Changes Since Submittal of the Licensee’s IPE

Since the licensee’s submittal of the CR-3 IPE, there have been several significant plant design changes incorporated into the PSA model that have resulted in a reduction in the CDF. The licensee’s summary of significant model changes incorporated due to these plant changes include:

- BEST added (“A” and “B” safeguards trains powered from separate transformers)
- FWP-7 (auxiliary feedwater pump) with dedicated diesel generator MTDG-1 installed
- Appendix R chiller installed
- EFP-3 (diesel-driven emergency feedwater pump) installed
- Low-pressure injection suction valves changed to be normally open
- High-pressure injection discharge throttle valves and cross-ties added
- Revision of emergency operating procedures reflected in human action probabilities

In addition to these plant changes, updates have been made to plant-specific data (through 1999) and initiating events data, as well as updates to the methods used for human reliability, common cause, internal flooding, and level two analyses.

The licensee states that, as of the date of the present submittal, there are no outstanding plant changes that would necessitate a change to the PSA model, and no planned plant changes that would be implemented prior to the fall 2003 refueling outage that would necessitate a change to the PSA model. Even if, for example, an additional EDG is acquired (or an additional AC source made available), the NRC staff finds the present risk impact results of the proposed 14-day EDG AOT to be low enough that no additional PSA modeling and computations need to be performed, including the additional EDG or AC source.

4.6 PSA Reviews

As discussed above, the original CR-3 PSA study was reviewed for the staff by Argonne National Laboratory as documented in NUREG/CR-5245. For the IPE submittal, multiple levels of review were used, including an assessment by Engineering and Operations personnel familiar with the plant design and operation. Subsequent revisions to the PSA models were performed by qualified individuals with knowledge of PSA methods and plant systems. Involvement by Engineering and Operations personnel, by providing input and reviewing results, was obtained when needed based on the scope of the changes being implemented.

The CR-3 PSA model and documentation was subjected to the industry peer certification review process in September 2001. In preparation for this review, an external consultant was hired to develop system notebook documentation. This required a review of the system models against plant drawings and procedures and identification of any inconsistencies with the models. Items identified from this review were considered and dispositioned. The internal flooding and common cause failures analyses were updated to current industry methodologies and data sources. An internal review of the PSA model elements and their corresponding documentation was conducted to assure that the model and documentation reflected the plant design.

The industry peer certification review was conducted by a diverse group of PSA engineers from other Babcock and Wilcox (B&W) plants, industry PSA consultants familiar with the B&W plant design, and a representative from the Institute of Nuclear Power Operations. The certification review covered all aspects of the PSA model and the administrative processes used to maintain and update the model. This review generated specific recommendations for model changes to correct errors, as well as guidance for improvements to processes and methodologies used in the CR-3 PSA model, and enhancements to the documentation of the model and the administrative procedures used for model updates.

Following completion of the previous review, the CR-3 PSA model was revised to address each issue identified that affected the model. The significant changes included:

- Update of plant-specific thermal-hydraulic analyses that provide the bases for accident sequences, systems success criteria, and timing for operator actions,

- Revision of accident sequence logic for steam generator tube rupture and anticipated transient without scram mitigation,
- Development of an initiating event to address the loss of all raw water pumps (loss of ultimate heat sink),
- Update of the interfacing systems' LOCA analyses,
- Update of the human reliability analysis, including the dependency analysis for multiple operator action responses to an event, and
- Update of the level two analysis.

Issues involving model documentation are being addressed as each individual PSA document is reviewed and approved under Progress Energy corporate procedures. Other changes involving guidance documents and administrative processes used for model updates are planned to be addressed by Progress Energy corporate procedures once the peer review process has been completed for all PSA models (including the Brunswick Nuclear Plant, the Robinson Nuclear Plant, and the Harris Nuclear Plant). According to the licensee, the issues identified by the peer review in these areas have been reviewed and determined not to have any impact on the present submittal. All other peer review items that impact the PSA model have been addressed and are reflected in the present submittal according to the licensee.

The NRC staff judges this licensee finding to be satisfactorily supportive of the present request.

4.7 External Events

4.7.1 Seismic

The licensee's IPEEE seismic evaluation was performed as an extension of the Unresolved Safety Issue A-46 Program. The screening review has identified no significant seismic vulnerabilities and has demonstrated that the plant safety-related structures, systems, and components are capable of withstanding the review-level earthquake designated for CR-3.

4.7.2 Fire

A review of the databases used for the IPEEE identified the fire zones that contained equipment that could, if damaged, disable the EDGs. Proposed TS Bases, Table B 3.8.1-1, lists these fire zones and the impacted EDGs. The presence of automatic suppression, fire wrap, and the total zone ignition frequency is also listed.

The CR-3 IPEEE fire assessment was performed by the application of fire-induced vulnerability evaluation methodology with PSA quantification of CDF estimates. Areas of strengths in the analysis include: (1) the development of an extensive database to tie circuits in various locations to specific components modeled in the IPE; (2) above-average documentation of the work performed; (3) thorough and focused walkdowns, including two to identify ignition sources; (4) consideration of recovery actions, including adjustment for fire events; and (5) thorough examination of fire ignition sources and their potential targets for unscreened fire zones. Other areas in the analysis do not impact the overall fire risk significantly.

As a means to provide additional protection for the EDG that is operable during maintenance on the opposite train, a periodic fire watch will be established by the licensee during extended EDG maintenance outages for fire areas that are considered risk-significant by the IPEEE, affect both EDGs, or have increased risk significance due to EDG maintenance.

4.7.3 Transportation and Nearby Facility Accidents

The licensee evaluated marine-related accidents, aircraft impact accidents, road and rail accidents, and fixed-facility accidents, including industrial and military facilities and pipelines. The licensee reported that no significant risk from these types of events was found. There are no major shipping lanes near the plant site. The Intercoastal Waterway is 10 miles offshore from the plant and has minimal marine traffic. No railroads are located within 5 miles. U.S. Route 19 passes within 4 miles of the site but has a relatively small volume of traffic and is not judged to contribute significantly to plant risk. The effect of aircraft crashes on the plant CDF was evaluated and was reported by the licensee to be extremely small (about 1E-7/r-yr). There are no commercial airports closer than 50 miles. There is a small airport within 8 miles, but the number of operations with small planes is too low to contribute any significant risk. The licensee evaluated accidents involving a fire or an explosion from nearby industrial facilities. The only significant facility within 5 miles of the site is a mining operation located about 4 miles from the plant. The licensee reported that there was sufficient distance between the mine and the plant site such that an explosion of up to 1000 lbs. of TNT would be insufficient to cause significant damage to the plant. The onsite fossil units are not risk-significant. The licensee reported that there are no hazardous pipelines passing on or near the plant site. The licensee also evaluated a release of toxic chemicals as a potential hazard. There were no facilities within 5 miles of the plant identified as using or storing toxic chemicals. The potential accidental releases of sulphur dioxide, ammonia, or chlorine were evaluated, and the licensee reported that control room habitability would not be impacted significantly by postulated releases of these chemicals at CR-3. The NRC staff finds that the licensee's discussion of transportation and nearby facility accidents is reasonable and conforms to the guidance of NUREG-1407.

4.7.4 Other External Events

CR-3 is located in an area subjected to frequent thunderstorms with associated lightning strikes. The licensee reported that modifications have been made in the switchyard loading configuration to reduce the likelihood of a LOOP due to a lightning strike. The licensee reported that, as a result of these modifications and accumulated plant experience, it has been concluded that lightning does not pose a significant risk. The licensee reported that there were no other risks identified at CR-3 that represent significant risks. The NRC staff agrees.

4.8 PSA Conclusions

The NRC staff concludes that the impact on plant risk of allowing 14 day at-power AOTs for the CR-3 EDGs is very small for both internal and external events. The NRC staff thus recommends that the proposed 14-day AOTs be approved.

5.0 NRC STAFF TECHNICAL CONCLUSION

The NRC staff has reviewed the proposed modification of the CR-3 TS to extend the AOT for the EDGs from 72 hours to 14 days and to modify two EDG SRs (SR 3.8.1.8 and SR 3.8.1.11) to allow performance of the SRs at power if the SRs are required to demonstrate EDG operability from a deterministic and a probabilistic risk assessment perspective. The NRC staff concludes from the deterministic and available risk insights and findings that the proposed changes do not affect CR-3's compliance with the intent of GDC 17; therefore, the proposed changes are acceptable.

6.0 STATE CONSULTATION

Based upon a letter dated March 8, 1991, from Ms. Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, to Ms. Deborah A. Miller, Licensing Assistant, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments. The State of Florida reconfirmed this position in a letter dated May 2, 2003.

7.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (67 FR 50955). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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