Mr. Charles M. Dugger
Vice President Operations
Entergy Operations, Inc.
17265 River Road
Killona, LA 70066-0751

May 25, 2000

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT RE: LOW PRESSURE SAFETY INJECTION ALLOWED OUTAGE TIME INCREASE (TAC NO. MA6311)

Dear Mr. Dugger:

The Commission has issued the enclosed Amendment No. 164 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated August 4, 1999, as supplemented by letter dated May 18, 2000.

The amendment modifies the TSs to extend allowed outage time (AOT) to seven days for one inoperable low pressure safety injection (LPSI) train. Additionally, an AOT of 72 hours is imposed for other conditions where the equivalent of 100 percent emergency core cooling system (ECCS) subsystem flow is available. If 100 percent ECCS flow is unavailable due to two inoperable LPSI trains, an ACTION has been added to restore at least one LPSI train to OPERABLE status within one hour or place the plant in HOT STANDBY in six hours, and to exit the MODE of applicability in the following six hours. In the event the equivalent of 100 percent ECCS subsystem flow is not available due to other conditions, TS 3.0.3 is entered. The Limiting Condition for Operation terminology is changed for consistency with the ECCS requirements and the associated TS Bases pages are changed.

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

Sincerely,
/RA/
N. Kalyanam, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures: 1. Amendment No. 164 to NPF-38
2. Safety Evaluation

cc w/encls: See next page

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

May 25, 2000

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17265 River Road
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 164
License No. NPF-38

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated August 4, 1999, as supplemented by letter dated May 18, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-38 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 164 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 25, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 164

TO FACILITY OPERATING LICENSE NO. NPF-38

DOCKET NO. 50-382

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

Remove	Insert
3/4 5-3	3/4 5-3
---------	3/4 5-3a
B 3/4 5-1b	B 3/4 5-1b
---------	B 3/4 5-1c
---------	B 3/4 5-1d
B 3/4 5-2	B 3/4 5-2

3/4.5.2 ECCS SUBSYSTEMS - MODES 1, 2, AND 3

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent emergency core cooling system (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection train,

b. One OPERABLE low-pressure safety injection train, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water storage pool on a safety injection actuation signal and automatically transferring suction to the safety injection system sump on a recirculation actuation signal.

APPLICABILITY: MODES 1, 2, and 3*#.

ACTION:

a. With one ECCS subsystem inoperable due to one low pressure safety injection train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia and RCS average temperature to less than 500°F within the following 6 hours.

b. With one or more ECCS subsystems inoperable due to conditions other than (a) and 100% of ECCS flow equivalent to a single OPERABLE ECCS subsystem available, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia and RCS average temperature to less than 500°F within the following 6 hours.

---

*With pressurizer pressure greater than or equal to 1750 psia.
#With RCS average temperature greater than or equal to 500°F.

LIMITING CONDITION FOR OPERATION

c. With both LPSI trains inoperable due to less than 100% of ECCS flow equivalent to a single OPERABLE ECCS subsystem, restore at least one LPSI train to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia and RCS average temperature to less than 500°F within the following 6 hours.

d. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

Each subsystem includes the piping, instruments, and controls to ensure the availability of an OPERABLE flowpath capable of taking suction from the RWSP on a SIAS and automatically transferring suction to the containment sump upon a recirculation actuation signal (RAS). The flowpath for each subsystem must maintain its designed independence to ensure that no single failure can disable both ECCS subsystems.

An ECCS subsystem is inoperable if it is not capable of delivering the design flow to the RCS. The individual components are inoperable if they are not capable of performing their automatic design function, or if supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent trains. Due to the redundancy of trains and the diversity of trains, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of these ACTIONs is to maintain a combination of OPERABLE equipment such that 100% of the ECCS flow equivalent to a single OPERABLE subsystem remains available.

100% of the ECCS flow equivalent to a single OPERABLE ECCS subsystem exists when the equivalent of one HPSI train, one LPSI train, and a suction flow path as described in the LCO are OPERABLE. The OPERABLE components may be in opposite subsystems. The HPSI component of the 100% ECCS flow equivalent may be composed of any combination of OPERABLE HPSI components such that flow is available to all four RCS loops. The LPSI component of the 100% ECCS flow equivalent may be composed of any combination of OPERABLE LPSI components such that flow is available to any two RCS loops. This allows increased flexibility in plant operations when components in opposite subsystems are inoperable.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS (Continued)

3.5.2, ACTION (a) addresses the specific condition where the only affected ECCS subsystem is a single LPSI train. A LPSI train consists of a pump, and two injection flow paths, including motor-operated valves operated by a common AC power source. The availability of at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS subsystem is implicit in the definition of ACTION (a).

If LCO 3.5.2 requirements are not met due to the condition described in ACTION (a), then the inoperable LPSI train components must be returned to OPERABLE status within seven (7) days of discovery. This seven (7) day Allowed Outage Time is based on the findings of deterministic and probabilistic analysis CE NPSD-995, "CEOG Joint Applications Report for Low Pressure Safety Injection System AOT Extension". Seven (7) days is a reasonable amount of time to perform many corrective and preventative maintenance items on the affected LPSI train. CE NPSD-995 concluded that the overall risk impact of the seven (7) day Allowed Outage Time was either risk-beneficial or risk-neutral.

ACTION (b) addresses other scenarios where the availability of at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS subsystem exists but the full requirements of LCO 3.5.2 are not met. If conditions of ACTION (b) were to exist, then inoperable components must be restored within 72 hours of discovery. The 72 hour Allowed Outage Time is based on an NRC reliability study (NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975) and is a reasonable amount of time to effect many repairs.

ACTION (c) addresses the condition in which 100% ECCS flow is unavailable due to two inoperable LPSI trains and requires restoration of at least one LPSI train to OPERABLE status within one hour or the plant placed in HOT STANDBY in 6 hours and reduce pressurizer pressure to less than 1750 psia and RCS average temperature to less than 500°F within the following 6 hours.

In the event less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS subsystem exists due to other conditions, LCO 3.0.3 is entered and the plant must be brought to a MODE (MODE 3 with pressurizer pressure less than 1750 psia and RCS average temperature less than 500°F) in which the LCO does not apply.

ECCS SUBSYSTEMS (Continued)

When in MODE 3 and with RCS temperature greater than or equal to 500°F two OPERABLE ECCS subsystems are required to ensure sufficient emergency core cooling capability is available to prevent the core from becoming critical during an uncontrolled cooldown (i.e., a steam line break) from greater than or equal to 500°F.

With the RCS temperature below 500°F and the RCS pressure below 1750 psia, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provides this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0. The requirement to dissolve a representative sample of TSP in a sample of water borated to be representative of post-LOCA sump conditions provides assurance that the stored TSP will dissolve in borated water at the postulated post-LOCA temperatures. A boron concentration of 3011 ppm boron is postulated to be representative of the highest post-LOCA sump boron concentration. Post LOCA sump pH will remain between 7.0 and 8.1 for the maximum (3011 ppm) and minimum (1504 ppm) boron concentrations calculated using the maximum and minimum post-LOCA sump volumes and conservatively assumed maximum and minimum source boron concentrations.

With the exception of systems in operation, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will prevent water hammer, pump cavitation, and pumping noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SIAS or during SDC. The 31 day frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the adequacy of the procedural controls governing system operation.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in

ECCS SUBSYSTEMS (Continued)

the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

The requirement to verify the minimum pump discharge pressure on recirculation flow ensures that the pump performance curve has not degraded below that used to show that the pump exceeds the design flow condition assumed in the safety analysis and is consistent with the requirements of ASME Section XI.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 164 TO

FACILITY OPERATING LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By application dated August 4, 1999, as supplemented by letter dated May 18, 2000, Entergy Operations, Inc., (EOI, the licensee, or Entergy) requested changes to the Technical Specifications (TS) for the Waterford Steam Electric Station, Unit 3 (Waterford 3). The supplement did not expand the scope of the application as noticed in the *Federal Register* and did not alter the proposed no significant hazards consideration determination.

The proposed amendment would allow extension of the allowed outage time (AOT) for one inoperable low pressure safety injection (LPSI) train from 72 hours to seven days. This will allow greater flexibility in the scheduling and implementation of maintenance on the subject equipment and avoid potential unscheduled plant shutdowns or requests for temporary relief for non-risk-significant conditions. Additionally, the end state for TS 3.5.2 is proposed to be changed to "reduce pressurizer pressure to less than 1750 psia and RCS [reactor coolant system] average temperature to less than 500 °F." This is consistent with the existing APPLICABILITY for TS 3.5.2, in MODE 3, with pressurizer pressure greater than or equal to 1750 psia and with RCS temperature greater than or equal to 500 °F.

2.0 BACKGROUND

Since the mid-1980's, the Nuclear Regulatory Commission (NRC) has been reviewing and granting improvements to TS that are based, at least in part, on probabilistic risk assessment (PRA) insights. In its final policy statement on TS improvements dated July 22, 1993 (58 FR 39132), the NRC stated that it...

> ...expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PSA [probabilistic safety assessment[1]] or risk survey and any available literature on risk insights and PSAs... Similarly, the NRC staff will also employ risk insights and PSAs in evaluating Technical Specifications related submittals. Further, as part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements.

The NRC reiterated this point when it issued the revision to 10 CFR 50.36, "Technical specifications," in July 1995. In August 1995, the NRC adopted a final policy statement on the use of PRA methods in nuclear regulatory activities that encouraged greater use of PRA to improve safety decision-making and regulatory efficiency. The PRA policy statement included the following points:

1. The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

2. PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements.

3. PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

In May 1995, the Combustion Engineering Owners Group (CEOG) submitted several Joint Application Reports for the staff's review. One of the CEOG Joint Application Reports provided justification for extension of the TS completion time for the LPSI system.[2] The justifications for this extension are based on a balance of probabilistic considerations, traditional engineering considerations, including defense-in-depth, and operating experience. Risk assessments for all of the Combustion Engineering (CE) plants are contained in the reports. The staff first reviewed the Joint Application Reports and then reviewed the licensee's plant-specific amendment request, which incorporated the Joint Application Reports by reference.

Arkansas Nuclear One, Unit 2 (ANO-2) had been the lead CE plant for the LPSI system TS changes. The staff performed an in-depth review of the ANO-2 PRA methodology related to these changes, as the lead plant for all of the CEOG. Therefore, a portion of the review of the Waterford 3 amendment request was based on a comparison of the Waterford 3 PRA results with those from ANO-2.

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[1]PSA and PRA are used interchangeably herein.

[2] CE NPSD-995, "Joint Application Report for Low Pressure Safety Injection System AOT Extension," May 1995.

3.0 EVALUATION

The staff evaluated the licensee's proposed amendment to extend the TS completion time (completion time and AOT are used interchangeably herein) for one LPSI train out of service from 72 hours to seven days using insights derived from traditional engineering considerations and the use of PRA methods to determine the safety impact of extending the completion times.

3.1 Traditional Engineering Evaluation

The current Waterford 3 TS addresses the LPSI system as a portion of the Emergency Core Cooling System (ECCS). The two trains of the LPSI system, in combination with the two trains of the high pressure safety injection (HPSI) system, form two redundant ECCS trains. TS 3.5.2 requires two ECCS trains to be operable. With one ECCS train inoperable, on the basis of any component inoperability but at least 100 percent of the ECCS flow equivalent to a single operable ECCS train available, the train must be returned to operable status within 72 hours or the plant must be placed in hot shutdown within the following six hours.

The proposed change will allow up to seven days for the licensee to restore operability to an inoperable LPSI train that is the cause of ECCS train inoperability. In some instances, corrective maintenance of the LPSI pump and valves and testing of valves may require taking one train of LPSI out of service for more than several days. Thus, repair within the existing completion time cannot be ensured and may result in an unscheduled shutdown or a request for temporary relief to allow continued plant operation. On the basis of the review of maintenance requirements of the LPSI train for CE pressurized water reactors (PWRs), the licensee determined that a seven-day completion time would provide sufficient margin to effect most anticipated, preventive, and corrective maintenance activities, and LPSI strain valve surveillance tests at power.

The LPSI trains, combined with the HPSI trains, form two redundant ECCS subsystems. The two LPSI pumps are high volume, low head centrifugal pumps designed to supplement the safety injection tank (SIT) inventory in reflooding the reactor vessel to ensure core cooling during the early stages of a large loss of coolant accident (LOCA).

The LPSI pumps take suction from the refueling water storage pool (RWSP) during the injection phase of a LOCA event and pump the water through two separate discharge headers. Prior to penetrating containment, each LPSI header splits into two injection paths, with individual injection valves. Each supply header has a motor operated flow control valve. Once inside containment, the LPSI headers combine with HPSI and SIT discharge piping and direct the flow through a common injection header into each of the four reactor coolant system cold legs. The LPSI system pumps start and valves open upon receipt of a safety injection actuation signal. When RWSP level is drawn down by inventory transfer during the injection phase, a low RWSP level actuates the recirculation actuation signal which stops the LPSI pumps and opens the Safety Injection System sump isolation valves. The HPSI pumps and containment spray pumps remain running for long-term containment and core cooling.

The LPSI system is also used in conjunction with a portion of the containment spray system for decay heat removal in the shutdown cooling alignment.

3.2 Probabilistic Risk Assessment Evaluation

The staff used a three-tiered approach to evaluate the risk associated with the proposed TS changes. The first tier evaluated the PRA model and the impact of the completion time extensions for the LPSI system on plant operational risk. The evaluation of the PRA model relied, in part, on a cross comparison approach with a similar plant. The second tier addressed the need to preclude potentially high risk configurations, by identifying the need for any additional constraints or compensatory actions that, if implemented, would avoid or reduce the probability of a risk-significant configuration during the time when one LPSI train is out of service. The third tier evaluated the licensee's Configuration Risk Management Program (CRMP) to ensure that the applicable plant configuration will be appropriately assessed from a risk perspective before entering into or during the proposed completion times. Based on discussions between NRC and Entergy, it was determined that the CRMP was not required to be in the TS and could be moved to a licensee-controlled program. This change is reflected in the resubmittal of the amendment package dated May 18, 2000. Each tier and the associated findings are discussed below.

3.2.1 Tier 1 Evaluation

After completing a detailed evaluation for the tentative approval of LPSI TS AOT extension for ANO-2, the original CEOG lead plant for the risk-informed TS pilot project, the staff used a cross comparison approach to consider the viability of similar AOT relaxations for other participating CEOG plants, including Waterford 3. The pilot technical evaluation report[3] used in support of the staff's draft safety evaluation for ANO-2[4] focused on:

- the process adopted by the CEOG to assess single AOT risk,
- the identification of ANO-2 accident sequences in which credit was taken for SITs and LPSI,
- independent verification of the single AOT risk [essentially equivalent to incremental conditional core damage probability (ICCDP)[5]], and
- determination of the significance of single AOT risk relative to an acceptance guideline value.

The objective of this cross comparison evaluation is to use insights derived from the ANO-2 technical evaluation to examine the validity of the conclusions drawn in the joint submittals. The staff believes that the findings of the lead pilot plant evaluation will be generally applicable to

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[3]SCIE-NRC-318-97, "Technical Evaluation of Combustion Engineering Owners Group (CEOG) Joint Application for Safety Injection Tanks and Low Pressure Safety Injection System Allowed Outage Time (AOT) Extension," July 21, 1997.

[4]SECY-97-095, "Probabilistic Risk Assessment Implementation Plan Pilot Application for Risk-Informed Technical Specifications," April 30, 1997.

[5]ICCDP = [(conditional Core Damage Frequency (CDF) with the subject equipment out of service) - (baseline CDF with nominal expected equipment unavailabilities)] X (duration of single AOT under consideration).

other CE plants, due to the fact that a common methodology was employed by the CEOG to quantify AOT risk, and CE plants have similar design characteristics. The staff confirmed that differences in the underlying PRA models are chiefly attributed to:

- minor design differences,
- operational differences,
- success criteria assumptions, and
- common cause failure β-factor or multiple Greek letter (MGL) assumptions.

The cross comparison draws on information contained in the CEOG Joint Application Reports, the licensees' responses to the staff's requests for additional information, the licensees' individual plant examinations (IPEs) performed in response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," and the corresponding IPE evaluations performed by the staff.

The following factors are chiefly responsible for the differences in LPSI AOT risks among the CE plants:

- use of LPSI to mitigate multiple initiating events,
- HPSI redundancies, and
- LPSI common cause β-factor or MGL assumptions

Based on the licensee's information in the CEOG November 1999 submittal, the staff estimates that the LPSI preventive and corrective maintenance weighted average single AOT risk for Waterford 3 is 5.3E-08 and is less than the acceptance guideline value 5.OE-07 from Regulatory Guide 1.177, "An Approach for Plant-Specific Risk-Informed Decisionmaking: Technical Specifications." Further, the staff feels that this estimate is reasonable since the conjoint frequency of large break LOCA, SIT malfunction, and deleterious break location is extremely small. In addition, the change in the Waterford 3 updated baseline CDF (as reported in CE NPSD-995, Revision 1), due to the LPSI AOT change, is about 0.6 percent, i.e., from 1.54E-05 per year to 1.55E-05 per year. The change in CDF of 1.OE-07 per year is within the acceptance guidelines published in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis."

The staff concludes that the approach and findings obtained for ANO-2, and the cross comparisons to other CE plants, are generally applicable to Waterford 3. To complete the first tier evaluation, the staff reviewed the quality of the Waterford 3 PRA.

Three levels of review were performed on the original Waterford 3 IPE submittal. The first was a basic Quality Assurance review carried out by the organization that developed the analysis. A qualified individual with knowledge of PSA methods and plant systems performed an independent review of all assumptions, calculations, and results for each task and the system models in the Level 1 analysis, performed with CAFTA/DOS software. Waterford 3 plant personnel not involved in the development of the PSA performed the second level of review. This review group consisted of individuals from Operations, Licensing, Engineering, and Training, providing diverse expertise with plant design and operations knowledge to review the system fault trees for accuracy. The third level of review was performed by PSA experts from

ERIN Engineering. ERIN provided broad insights on techniques and results based on experience from other plant PSAs. They reviewed the overall PSA methodology, accident sequence analyses, system fault trees, Level 1 results, and the human failure and recovery analysis. The licensee uses an Institute of Nuclear Power Operations-accredited training program for PSA personnel.

The Waterford 3 PSA model has been updated with CAFTA/CQUANT 32 software since the development of the IPE in accordance with the "living model" philosophy at Waterford 3 and in the industry. The Waterford 3 IPE is considered to be Revision 0 of the Waterford 3 PSA model. The model is currently at the Revision 2, Change 1 stage. Some of the major changes that have been incorporated since the IPE submittal are as follows: the elimination of asymmetries across multiple train systems (allowing the swing trains to recover either A or B trains, rather than only one), the inclusion of additional DC power dependencies on applicable systems, the incorporation of a detailed convolution methodology of calculating offsite power recovery factors, and the update of some failure rate data. Also included were some minor changes that have occurred to the plant since the IPE submittal, such as the enhancement of certain simplified assumptions and the correction of minor errors found over the years (e.g., mis-classification of a valve as a motor-operated valve instead of an air-operated valve, or basic event description changes).

Since the IPE, every change to the PSA model has been prepared by one of the Waterford 3 PSA engineers; reviewed by a separate, independent PSA engineer; and approved by the Manager, Safety and Engineering Analysis.

A cross comparison of the Waterford 3 risk-related results that support the LPSI AOT extension was made with the other CEOG plants, as part of the generic CE-NPSD-995, Revision 1, "Joint Application Report for Low Pressure Safety Injection System AOT Extension." This provided another level of review for the Waterford 3 results.

During the week of January 17, 2000, a PSA Certification Team reviewed the Waterford 3 PSA Model. The certification was scheduled through CEOG participation. The team was made up of a lead from CE and four experienced PSA peers from other CE plants. The team identified some concerns, most of which had been previously identified by Entergy personnel. The team also identified some conservatisms. Entergy will develop a plan to prioritize all of the PSA Certification Team's concerns and implement the necessary improvements. Assurance that changes to the as-built and as-operated condition of the plant are incorporated into the PSA model is provided by the required review of all designed changes by the Safety and Engineering Analysis Group. This allows design changes to be screened for impact on the model.

When CRMP implementation is completed, a documented methodology for PSA update (based on the existing site calculation procedure) will be instituted. This will proceduralize a consistent, repeatable methodology for model update, and a consistent reflection of plant and operating changes. It also provides guidance on PSA applications, which may need to be re-reviewed for impact after updates, such as AOT extension inputs. In addition, incorporation of PSA-related questions on the screening checklists located in the Engineering Request and Procedure Development Procedures is being considered. These screening questions will trigger the preparer to have a PSA review for any change that may affect the as-built or as-operated condition of the plant.

The staff finds that the small ICCDP estimated for the change in AOT from three to seven days is consistent with the credit taken for the system in the PRA modeling, and that the extensive licensee review of the PRA models provides reasonable assurance that the models appropriately reflect the equipment and procedural characteristics at the plant.

This completes the staff's first tier evaluation of the licensee's proposal to extend the completion time for one LPSI train from three to seven days. Based on the above discussion, the staff finds acceptable the PRA model used by the Waterford 3 licensee and also concludes that there is minimal impact on the completion time extensions for the LPSI system on plant operational risk.

3.2.2 Tier 2 Evaluation

The licensee did not identify any dominant, risk-significant configurations associated with the proposed LPSI train completion time extension. The licensee concurs with the CEOG finding that a review of large, early release scenarios for the CE PWRs indicates that early releases arise as a result of the following class scenarios:

1. Containment Bypass Events

   These events include interfacing system LOCAs and steam generator (SG) tube ruptures with a concomitant loss of SG isolation (e.g., stuck open main steam safety valve).

2. Severe Accidents Accompanied by Loss of Containment Isolation

   These events include any severe accident in conjunction with an initially unisolated containment.

3. Containment Failure Associated with Energetic Events in the Containment

   Events causing containment failure included those associated with the High-Pressure Melt Ejection phenomena (including direct containment heating and hydrogen conflagrations/detonations.

Of the three radioactive release categories associated with the above event categories, Class 1 tends to represent a large, early release of potentially direct, unscrubbed fission products, to the environment. Class 2 events encompass a range of releases, varying from early to late, that may or may not be scrubbed. Class 3 events result in a high-pressure failure of the containment, typically immediately upon or slightly after reactor vessel failure. Detailed Level 2 analysis for the plant condition with one LPSI train inoperable was not performed.

1. Containment Bypass Events

Events contained in this category that may rely on the LPSI for event mitigation include the large Interfacing System LOCA (ISLOCA), i.e., failure of a shutdown cooling line. Testing and/or maintenance of containment insolation valves residing in the LPSI system are governed under the plant TS. Thus, no change in the ISLOCA frequency is expected.

ISLOCAs are characterized by continuous and unreplenished loss of RCS inventory and makeup. In these scenarios, core damage ultimately results following the depletion of reactor coolant. Thus, provided that a continuous, independent water supply is not available during the accident, the ISLOCA will progress into early core damage regardless of LPSI availability.

2. Severe Accidents Accompanied by Loss of Containment Isolation

Another event contributing to large, early fission product releases could occur when an unmitigated large LOCA occurs in conjunction with an initially unisolated containment. Significant fission product releases would not occur unless the containment atmosphere is unscrubbed, i.e., sprays are inoperable. This latter combination of events is considered of very low probability and would not significantly increase with a decrease in LPSI pump availability, because LPSI is not a major support system for the containment sprays.

3. Containment Failure Associated with Energetic Events in the Containment

Class 3 events are dominated by RCS transients that occur at high pressure. These events exclude those where LPSI system performance would be called for and, therefore, LPSI status is not a contributor to this event category. It is, therefore, concluded that increased unavailability of the LPSI system (as could potentially result as a consequence of an increased AOT) will have a very small impact on the large, early release fraction for CE PWRs.

External events can potentially lead to high risk configurations and, therefore, are included in the second tier evaluation. The LPSI at-power function is largely to mitigate large LOCA events.

The external events of fire, severe weather, and flooding are not considered to be initiators of large LOCA events. The only external events that need to be considered are seismic events. The IPE of externally initiated events-seismic events evaluation, however, showed that there are no seismic vulnerabilities.

Thus, the staff concurs that a large LOCA is not considered to be a credible consequence of a seismic event for Waterford 3, and that extending the completion time for a LPSI train will not increase plant risk where external phenomena are the initiating events.

The Tier 2 evaluation did not identify the need for any additional constraints or compensatory actions that, if implemented, would avoid or reduce the probability of a risk-significant configuration.

3.2.3 Tier 3 Evaluation

The licensee proposes to implement a CRMP and to establish the CRMP requirements in a licensee-controlled document. The purpose of the CRMP is to ensure that a proceduralized PRA-informed process is in place that assesses the overall impact of plant maintenance on plant risk.

Implementation of the CRMP will enable appropriate actions to be taken or decisions to be made to minimize and control risk when performing on-line maintenance for systems, structures, and components (SSCs) with a risk-informed completion time.

The scope of the SSCs included in the CRMP are those SSCs modeled in the licensee's plant PRA in addition to those SSCs considered of High Safety Significance per Regulatory Guide 1.160, Revision 2 (the Maintenance Rule regulatory guide), that are not modeled in the PRA.

The content of the CRMP process consists of the following components:

1. Provisions for the control and implementation of a Level 1 at-power internal events PRA-informed methodology. The assessment is to be capable of evaluating the applicable plant configuration.

2. Provisions for performing an assessment prior to entering the plant configuration described by the LCO Action Statement for preplanned activities.

3. Provisions for performing an assessment after entering the plant configuration described by the LCO Action Statement for unplanned entry into the LCO Action Statement.

4. Provisions for assessing the need for additional actions after the discovery of additional equipment-out-of service conditions while in the plant configuration described by the LCO Action Statement.

5. Provisions for considering other applicable risk-significant contributors such as Level 2 issues and external events, qualitatively or quantitatively.

Key Element 1. Implementation of CRMP

The intent of the CRMP is to implement subsection (a)(3) of the Maintenance Rule (10 CFR 50.65) with respect to on-line maintenance for risk-informed technical specifications, with the following additions and clarifications:

1. The scope of the SSCs to be included in the CRMP will be those SSCs modeled in the licensee's plant PRA in addition to those SSCs considered of High Safety Significance per Regulatory Guide 1.160, Revision 2 (the Maintenance Rule regulatory guide), that are not modeled in the PRA.

2. The CRMP assessment tool is PRA informed, and may be in the form of either a risk matrix, an on-line assessment, or a direct PRA assessment.

3. The CRMP will be invoked as follows for:

   Risk-Informed Inoperability: A risk assessment will be performed prior to entering the LCO Condition for preplanned activities. For unplanned entry into the LCO Condition, a risk assessment will be performed in a time frame consistent with the plant's Corrective Action Program.

   Additional SSC Inoperability and/or Loss of Functionality: When in the risk-informed Completion Time, if an additional SSC within the scope of the CRMP becomes inoperable/non-functional, a risk assessment shall be performed in a time frame consistent with the plant's Corrective Action Program.

4. Tier 2 commitments apply for planned maintenance only, but will be evaluated as part of the Tier 3 assessment for unplanned occurrences.

Key Element 2. Control and Use of the CRMP Assessment Tool

1. Plant modifications and procedure changes will be monitored, assessed, and dispositioned.

   - Evaluation of changes in plant configuration or PRA model features can be dispositioned by implementing PRA model changes or by the qualitative assessment of the impact of the changes on the CRMP assessment tool. This qualitative assessment recognizes that changes to the PRA take time to implement and that changes can be effectively compensated for without compromising the ability to make sound engineering judgments.

   - Limitations of the CRMP assessment tool are identified and understood for each specific Completion Time extension.

2. Procedures exist for the control and application of CRMP assessment tools, including description of the process when outside the scope of the CRMP assessment tool.

Key Element 3. Level 1 Risk-Informed Assessment

The CRMP assessment tool is based on a Level 1, at power, internal events PRA model. The CRMP assessment may use any combination of quantitative and qualitative input. Quantitative assessments can include reference to a risk matrix, pre-existing calculations, or new PRA analyses.

1. Quantitative assessment should be performed whenever necessary for sound decision making.

2. When quantitative assessments are not necessary for sound decision making, qualitative assessments will be performed. Qualitative assessments will consider applicable, existing insights from quantitative assessments previously performed.

Key Element 4. Level 2 Issues/External Events

External events and Level 2 issues are treated qualitatively and/or quantitatively.

Guidance for implementing the CRMP is provided by plant procedures.

The licensee also has the ability to analyze the risk impact of outage configurations in a timely manner using a tool called the Equipment-out-of-Service (EOOS) software.

The staff's third tier evaluation concludes that the risk-informed CRMP proposed by the licensee will satisfactorily assess the risk associated with the removal of equipment from service during the proposed LPSI AOT. The program provides the necessary assurances that appropriate assessments of plant risk configurations, including during outage conditions, are sufficient to support the completion time extension request for the LPSI system.

3.3 Summary

The staff has evaluated the licensee's proposed changes for compliance with regulatory requirements, as documented in this evaluation, and has determined that they are acceptable. This determination is based on the following:

1. The traditional engineering evaluation reveals that increasing the availability of the LPSI system for shutdown cooling during outages by performing preventive and corrective maintenance at power can contribute to an overall enhancement of plant safety.

2. The staff finds the PRA model used by the Waterford 3 licensee acceptable and also concludes that there is minimal impact of the completion time extensions for the LPSI system on plant operational risk (Tier 1 evaluation).

3. The review of potentially high risk configurations did not identify the need for any additional constraints or compensatory actions that, if implemented, would avoid or reduce the probability of a risk-significant configuration (Tier 2 evaluation).

The risk-informed CRMP proposed by the licensee will satisfactorily assess the risk associated with the removal of equipment from service during the proposed LPSI AOT (Tier 3 evaluation) and will be managed by plant procedures.

The staff therefore, finds that the completion time for one LPSI train may be extended to seven days, with a negligible impact on risk. Additionally, the staff finds acceptable the change of end state for TS 3.5.2 to "reduce pressurizer pressure to less than 1750 psia and RCS average temperature to less than 500 °F" for consistency with the applicability for TS 3.5.2.

4.0 <u>STATE CONSULTATION</u>

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (65 FR 4278, dated January 26, 2000). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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