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

May 15, 2000

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT RE: EXTENDING THE ALLOWABLE OUTAGE TIME FROM 72 HOURS TO SEVEN DAYS FOR ONE INOPERABLE CONTAINMENT SPRAY SYSTEM (TAC NO. MA6177)

Dear Mr. Dugger:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 163 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3. The amendment consists of changes to the plant technical specifications (TS) in response to your application dated July 29, 1999.

The amendment modifies TS 3.6.2.1 to extend the allowable outage time to seven days for one containment spray system (CSS) train inoperable. A new ACTION has been added to provide a shutdown requirement for the inoperability of two CSSs. The associated changes to TS Bases are included. Your request that the APPLICABILITY of TS 3.6.2.1 be changed to an end state of MODE 4 is not found acceptable at this time, since a generic industry initiative on this subject is presently being pursued.

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. 163 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 15, 2000

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Vice President Operations
Entergy Operations, Inc.
17265 River Road
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N. Kalyanam, Project Manager, Section 1
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Docket No. 50-382

Enclosures: 1. Amendment No. 163 to NPF-38
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cc w/encls: See next page

Waterford Generating Station 3

cc:

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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. 163
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 July 29, 1999, 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) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 163, 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 90 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 15, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 163

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 6-16	3/4 6-16
3/4 6-17	3/4 6-17
B 3/4 6-3	B 3/4 6-3
B 3/4 6-4	B 3/4 6-4
B 3/4 6-4a	------------
B 3/4 6-5	B 3/4 6-5
------------	B 3/4 6-6
------------	B 3/4 6-7

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWSP on a containment spray actuation signal and automatically transferring suction to the safety injection system sump on a recirculation actuation signal. Each spray system flow path from the safety injection system sump shall be via an OPERABLE shutdown cooling heat exchanger.

APPLICABILITY: MODES 1, 2, 3 and 4*.

ACTION:

a. With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

b. With two containment spray systems inoperable, restore at least one spray system to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the water level in the containment spray header riser is > 149.5 feet MSL elevation.

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is correctly positioned to take suction from the RWSP.

c. By verifying, that on recirculation flow, each pump develops a total head of greater than or equal to 219 psid when tested pursuant to Specification 4.0.5.

*With Reactor Coolant System Pressure > 400 psia.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months, during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a CSAS test signal.

   2. Verifying that upon a recirculation actuation test signal, the safety injection system sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.

   3. Verifying that each spray pump starts automatically on a CSAS test signal.

e. At least once per 10 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM (Continued)

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 La leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

Operability concerns for purge supply and exhaust isolation valves other than those addressed in Actions "a" and "b" of Specification 3.6.1.7 are addressed under Specification 3.6.3, "Containment Isolation Valves."

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT SPRAY SYSTEM and CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Spray System and the Containment Cooling System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or MSLB for any double-ended break of the largest reactor coolant pipe or main steam line. Under post-accident conditions these systems will maintain the containment pressure below 44 psig and temperatures below 269.3°F during LOCA conditions or 413.5°F during MSLB conditions. The systems also reduce the containment pressure by a factor of 2 from its post-accident peak within 24 hours, resulting in lower containment leakage rates and lower offsite dose rates.

The Containment Spray System (CSS) also provides a mechanism for removing iodine from the containment atmosphere under post-LOCA conditions to maintain doses in accordance with 10 CFR Part 100 limits as described in Section 6.5.2 of the FSAR.

If LCO 3.6.2.1 requirements are not met due to the condition described in ACTION (a), then the inoperable CSS 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-1045, "Modifications To The Containment Spray System, and Low Pressure Safety Injection System Technical Specifications". Seven (7) days is a reasonable amount of time to perform many corrective and preventative maintenance items on the affected CSS train. CE NPSD-1045 concluded that the overall risk impact of the seven (7) day allowed outage time was either risk-beneficial or risk-neutral.

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT SPRAY SYSTEM and CONTAINMENT COOLING SYSTEM (Continued)

ACTION (b) addresses the condition in which two CSS trains are inoperable and requires restoration of at least one spray system to OPERABLE status within 1 hour or the plant to be placed in HOT STANDBY in 6 hours and COLD SHUTDOWN within the following 30 hours. (COLD SHUTDOWN is the acceptable end state.)

In MODE 4 when shutdown cooling is placed in operation, the Containment Spray System is realigned in order to allow isolation of the spray headers. This is necessary to avoid a single failure of the spray header isolation valve causing Reactor Coolant System depressurization and inadvertent spraying of the containment. To allow for this realignment, the Containment Spray System may be taken out-of-service when RCS pressure is ≤ 400 psia. At this reduced RCS pressure and the reduced temperature associated with entry into MODE 4, the probability and consequences of a LOCA or MSLB are greatly reduced. The Containment Cooling System is required OPERABLE in MODE 4 and is available to provide depressurization and cooling capability.

A train of Containment Cooling consists of two fans (powered from the same safety bus) and their associated coolers (supplied from the same cooling water loop). One Containment Cooling train and Containment Spray train has sufficient capacity to meet post accident heat removal requirements.

Operating each containment cooling train fan unit for 15 minutes and verifying a cooling water flow rate of 625 gpm ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected and corrective action taken.

The 18 month Surveillance Requirement verifies that each containment cooling fan actuates upon receipt of an actual or simulated SIAS actuation signal. The 18 month frequency is based on engineering judgment and has been shown to be acceptable through operating experience.

Verifying a cooling water flow rate of 1200 gpm to each cooling unit provides assurance that the design flow rate assumed in the safety analyses will be achieved. The safety analyses assumed a cooling water flow rate of 1100 gpm. The 1200 gpm requirement accounts for measurement instrument uncertainties and potential flow degradation. Also considered in selecting the 18 month frequency were the known reliability of the Cooling Water System, the two train redundancy, and the low probability of a significant degradation of flow occurring between surveillances. The flow measurement for the 18 month test shall be done in a configuration equivalent to the accident lineup to ensure that in an accident situation adequate flow will be provided to the containment fan coolers for them to perform their safety function.

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT SPRAY SYSTEM and CONTAINMENT COOLING SYSTEM (Continued)

Verifying that each valve actuates to the full open position provides further assurance that the valves will travel to their full open position on a Safety Injection Actuation Signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through GDC 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

"Containment Isolation Valves", previously Table 3.6-2, have been incorporated into the Technical Requirements Manual (TRM).

For penetrations with multiple flow paths, only the affected flow path(s) is required to be isolated when a containment isolation valve in that flow path is inoperable. The flow path may be isolated with the inoperable valve in accordance with the Action requirements, provided the leakage rate acceptance criteria, as applicable, is met and controls are in place to ensure the valve is closed. Also, the penetration is required to meed the requirements of GDC-54, and GDC-55 through GDC 57, as applicable, for all the unisolated flow paths.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

SURVEILLANCE REQUIREMENT SR 4.6.4.2.a requires performance of a system functional test for each hydrogen recombiner to ensure that the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR requires verification that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes. After reaching 700°F, the power is increased to maximum for approximately 2 minutes and verified to be ≥ 60 kW.

SURVEILLANCE REQUIREMENT SR 4.6.4.2.b ensures that there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failures involve loss of power, blockage of the internal flow path, missile impact, etc. A visual inspection is sufficient to determine abnormal conditions that could cause such failures.

SURVEILLANCE REQUIREMENT SR 4.6.4.2.c requires performance of a resistance to ground test for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms.

3/4.6.5 VACUUM RELIEF VALVES

The OPERABILITY of the primary containment to annulus vacuum relief valves with a setpoint of less than or equal + 0.3 psid ensures that the containment internal pressure differential does not become more negative than the containment design limit for internal pressure differential of 0.65 psi. This situation would occur, for the worst case, if all containment heat removal systems (containment spray, containment cooling, and other HVAC systems) were inadvertently started with only one vacuum relief valve OPERABLE.

3/4.6.6 SECONDARY CONTAINMENT

3/4.6.6.1 SHIELD BUILDING VENTILATION SYSTEM

The OPERABILITY of the shield building ventilation systems ensures that containment vessel leakage occurring during LOCA conditions into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. This requirement is necessary to meet the assumptions used in the safety analyses and limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during LOCA conditions.

Operation of the system with the heaters on for at least 10 hours continuous over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. Obtaining and analyzing charcoal samples after 720 hours of adsorber operation (since the last sample and analysis) ensures that the adsorber maintains the efficiency assumed in the safety analyses and is consistent with Regulatory Guide 1.52.

3/4.6.6.2 SHIELD BUILDING INTEGRITY

SHIELD BUILDING INTEGRITY ensures that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with operation of the shield building ventilation system, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.6.3 SHIELD BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment shield building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide (1) protection for the steel vessel from external missiles, (2) radiation shielding in the event of a LOCA, and (3) an annulus surrounding the steel vessel that can be maintained at a negative pressure during accident conditions. A visual inspection is sufficient to demonstrate this capability.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 163 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 July 29, 1999, 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 proposed amendment would allow extension of the allowed outage time (AOT) for one containment spray system (CSS) train inoperable from 72 hours to seven days. This AOT is controlled by TS 3.6.2.1. 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 licensee proposed to change the applicability for TS 3.6.2.1 to provide an end state of MODE 4. TS 3.6.2.1 currently provides applicability for MODES 1, 2, 3, and 4 (with reactor coolant system (RCS) pressure > 400 pounds per square inch, absolute in MODE 4). Action on this is being deferred pending the outcome of an industry initiative in this area.

2.0 BACKGROUND

Since the mid-1980's, the Nuclear Regulatory Commission (NRC or the Commission) 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

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[1]PSA and PRA are used interchangeably herein.

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 March 1998, the Combustion Engineering Owners Group (CEOG) submitted a Joint Application Report for the staff's review. It provided justifications for the extension of the TS AOT for the CSS system.[2] The justifications for this extension are based on a balance of probabilistic considerations and 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 Report and then reviewed the licensee's plant-specific amendment request, which incorporated the Joint Application Report by reference.

3.0 EVALUATION

The staff evaluated the licensee's proposed amendment to extend the TS AOT for one CSS 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 AOTs.

3.1 Traditional Engineering Evaluation

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[2]CE NPSD-1045, "Joint Applications Report for Modifications to the Containment Spray System, and Low Pressure Safety Injection System Technical Specifications," March 1998

The function of the containment heat removal systems under accident conditions is to remove heat from the containment atmosphere, thus maintaining the containment pressure and temperature at acceptably low levels. The systems also serve to limit offsite radiation levels by reducing the pressure differential between the containment atmosphere and the external environment, thereby decreasing the driving force for fission product leakage across the containment. The two containment heat removal systems are the containment cooling system (CCS) and the CSS. The CCS fan coolers are designed to operate during both normal plant operations and under loss-of-coolant accident (LOCA) or main steam line break (MSLB) conditions. The CSS is designed to operate during accident conditions only.

The heat removal capacity of the CCS and CSS is sufficient to keep the containment temperature and pressure below design conditions for any size break, up to and including double ended break of the largest reactor coolant pipe. The systems are also designed to mitigate the consequences of any size break, up to and including a double-ended break of a main steam line. The CCS and CSS continue to reduce containment pressure and temperature and maintain them at acceptable levels post-accident.

The CCS and CSS each consist of two redundant loops and are designed such that a single failure does not degrade their ability to provide the required heat removal capability. Two of four containment fan coolers and one CSS loop are powered from one safety-related bus. The other two containment fan coolers and CSS loop are powered from another independent safety related bus. The loss of one bus does not affect the ability of the containment heat removal systems to maintain containment temperature and pressure below the design values in a post-accident mode.

The CSS consists of two independent and redundant loops each containing a spray pump, shutdown heat exchanger, piping, valves, spray headers, and spray nozzles. It has two modes of operation, which are:

1. The injection mode, during which the system sprays borated water from the refueling water storage pool (RWSP) into the containment, and

2. The recirculation mode, which is automatically initiated by the recirculation actuation signal (RAS) after low level is reached in the RWSP. During this mode of operation, the safety injection system (SIS) sump provides suction for the spray pumps.

Containment spray is automatically initiated by the containment spray actuation signal coincident with the safety injection actuation signal and high containment pressure signal. If required, the operator can manually activate the system from the main control room.

Each CSS pump, together with a CCS loop, provides the flow necessary to remove the heat generated inside the containment following a LOCA or MSLB. Upon system activation, the pumps are started and the borated water flows into the containment spray headers.

When low level is reached in the RWSP, sufficient water has been transferred to the containment to allow for the recirculation mode of operation. Spray pump suction is automatically realigned to the SIS sump upon an RAS.

During the recirculation mode, the spray water is cooled by the shutdown heat exchangers prior to discharge into the containment. The shutdown heat exchangers are cooled by the component cooling water system.

Post-LOCA pH control is provided by trisodium phosphate dodecahydrate, which is stored in stainless steel baskets located in the containment near the SIS sump intake.

Based on a review of the design basis requirements for the CSS, the staff concluded that the loss of one CSS train is well within the design basis analyses and extending the AOT for the loss of one train from 72 hours to seven days may actually provide an overall safety benefit by avoiding potential unscheduled plant shutdowns for non-risk-significant conditions.

The plant status, with both CSS inoperable, is covered by TS 3.6.2.1, ACTION b., which has been added to say:

> With two containment spray systems inoperable, restore at least one spray system to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours.

ACTION b. addresses the condition in which two CSS trains are inoperable and requires restoration of at least one spray system to OPERABLE status within 1 hour or the plant to be placed in HOT SHUTDOWN in 6 hours and COLD SHUTDOWN within the following 30 hours, with COLD SHUTDOWN being the acceptable end state. These requirements are consistent with similar requirements elsewhere in the TS and therefore acceptable.

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 AOT extension for the CSS on plant operational risk. The evaluation of the PRA model relied, in part, on a cross comparison approach with similar plants. 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 CSS train is out of service. The third tier evaluated the licensee's proposed 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 AOTs. However, 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 document.

Each tier and the associated findings are discussed below.

3.2.1 Tier 1 Evaluation and PRA Quality Review

The staff used a cross comparison approach to consider the viability of similar AOT relaxations for participating CEOG plants, including Waterford 3. The staff's safety evaluation focused on:

- the process adopted by the CEOG to assess single AOT risk,
- independent verification of the single AOT risk (essentially equivalent to incremental conditional core damage probability (ICCDP)[3]), 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 derived insights to examine the validity of the conclusions drawn in the joint submittals. The staff believes that the findings of a plant evaluation will be generally applicable to 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
- common cause failure β-factor or multiple Greek letter (MGL) assumptions
- non-presence of fan coolers (Palo Verde only)
- non-crediting of fan coolers (Arkansas Nuclear One, Unit 2 only)

The cross comparison draws on information contained in the CEOG Joint Application Report, 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 of External Events (IPEEE) for Severe Accident Vulnerabilities," and the corresponding IPE evaluations performed by the staff.

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

- non-presence of fan coolers (Palo Verde only)
- non-crediting of fan coolers
- CSS common cause β-factor or MGL assumptions

The effect of removing a train of the CSS on the ability of the subject CE plants to mitigate the consequences of core damage, in part, is measured by change in large early release frequency (ΔLERF) or by incremental conditional large early release probability (ICLERP),
where ICLERP = (ΔLERF-LERF)X(duration of single AOT under consideration).

The guidance measure for ICLERP is 5.0E-8. Specifically, the Regulatory Guide 1.177 states:

[3]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).

"The licensee has demonstrated that the TS AOT change has only a small quantitative impact on plant risk. An ICCDP[4] of less than 5.0E-7 is considered small for a single TS AOT change[5]. An ICLERP of 5.0E-8 or less is also considered small. Also, the ICCDP contribution should be distributed in time such that any increase in the associated conditional risk is small and within the normal operating background (risk fluctuations) of the plant (Tier 1)."

Based on the licensee's information in the March 1998 CEOG submittal, the CSS preventive and corrective maintenance weighted average single AOT risk for Waterford 3 is 0.0 and is less than the acceptance guideline value 5.0E-07 from Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications." In addition, the change in the Waterford 3 updated baseline CDF (as reported in CE NPSD-1045) due to the CSS AOT change is about 0.0 percent, (i.e., from 1.63E-05 per year to 1.63E-05 per year). The estimated null change in CDF is within the acceptance guidelines published in Regulatory Guide 1.177, and is due to the fact that the CSS does not provide any core heat removal or RCS inventory makeup function. Therefore, CSS availability does not impact CDF.

Additionally, in the CEOG's March 15, 1999, response to the staff RAI, it was determined that, for CE plants with diverse containment heat removal (i.e., containment sprays and fan coolers), the ICLERP is less than 3.0E-09. Waterford 3 is such a plant, and thus its ICLERP is less than the acceptance guideline value 5.0E-08 from Regulatory Guide 1.177.

The staff concludes that the cross comparisons to other CE plants in CE NPSD-1045 support the risk analysis and findings for 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). ERIN provided broad insights on techniques and results based on

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

[5]The ICCDP acceptance guideline of 5.0E-7 is based upon the hypothetical situation in which the subject equipment at a representative plant is out for five hours, causing the CDEF of the plant, with an assumed baseline CDF of 1.0E-4 per reactor year, to conditionally increase to 1.0E-3 per reactor year during the five-hour period. This basis assumes that the majority of repairs can be made in five hours or less and that the NRC has accepted this level of risk for existing operating plants.

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 CSS AOT extension was made with the other CE plants, as part of the generic evaluation in CE NPSD-1045. 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 the 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, mainly involving non-modeling of air-operated valve common cause failure. The licensee has determined that this is not a meaningful contributor to ICCDP or ICLERP. 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 design changes by the Safety and Engineering Analysis Group. This allows design changes to be screened for impact on the model.

When the licensee's 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 and ICLERP estimated for the change in AOT from 72 hours 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 AOT for one CSS train from 72 hours 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 of the AOT extensions for the CSS 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 CSS train AOT extension.

3.2.3 Tier 3 Evaluation

The licensee proposes to implement a CRMP and to establish the CRMP requirements in the Waterford 3 Site Directive. 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 AOT.

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 Limiting Conditions for Operation (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 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 Action Statement for preplanned activities. For unplanned entry into the LCO Action Statement, 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 AOT, 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 AOT 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 software (EOOS).

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 CSS AOT. The program provides the necessary assurances that appropriate assessments of plant risk configurations, including during outage conditions, are sufficient to support the AOT extension request for the CSS 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 the loss of one CSS train is well within the design basis analyses and extending the AOT for the loss of one train from 72 hours to seven days may actually provide an overall safety benefit in some cases by avoiding potential unscheduled plant shutdowns for non-risk-significant conditions.

2. The staff finds acceptable the PRA model used by the Waterford 3 licensee and also concludes that there is minimal impact of the AOT extensions for the CSS 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 CSS AOT (Tier 3 evaluation) and will be managed by plant procedures.

The staff therefore, finds that the AOT for one CSS train may be extended to seven days, with a negligible impact on risk. However, the staff does not find acceptable at this time, a MODE 4 end state for TS 3.6.2.1, since a generic industry initiative on this subject is presently being pursued.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Louisiana State official was notified of the proposed issuance of the amendments. 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 (64 FR 6406, dated February 9, 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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