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

September 5, 2000

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

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT RE: ADDITION OF MAIN FEEDWATER ISOLATION VALVES TO TECHNICAL SPECIFICATIONS AND REQUEST FOR NRC STAFF REVIEW OF AN UNREVIEWED SAFETY QUESTION (TAC NO. MA6173)

Dear Mr. Dugger:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 167 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 July 15, 1999 (NPF-38-217), as supplemented by letter dated March 29, 2000.

The amendment creates a new TS for the Main Feedwater Isolation Valves (MFIV) Section modeled after the guidelines of TS 3.7.3 in NUREG-1432, "Standard Technical Specifications - Combustion Engineering Plants." Additionally, the letter provides for the NRC staff review of an unreviewed safety question (USQ) regarding the crediting of the Reactor Trip Override feature and Auxiliary Feedwater Pump high discharge pressure trip as assisting the operation of the MFIVs during their required safety function, to close on a Main Steam Isolation Signal.

By approving this amendment, the staff is not removing the licensing basis requirement that only safety related equipment be used to evaluate the plant response to design basis accidents or changing the valve response requirement of closure within five seconds.

A revision to the Final Safety Analysis Report will be made to address the changes associated with this TS Change Request and the resolution of the USQ.

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. 167 to NPF-38
2. Safety Evaluation

cc w/encls: See next page

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DATE	7/17/00	7/17/00	8/29/00	8/21/00	8/29/00

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~~* No major changes made in the SE input dated 7/7/2000.~~

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,

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. 167 to NPF-38
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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. 167
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 15, 1999, as supplemented by letter dated March 29, 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. 167, 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.

   Additionally, the license is amended to approve changes to the Final Safety Analysis Report made to address the changes associated with this TS Change Request and the resolution of the unreviewed safety question. Entergy Opeations, Inc. shall update the FSAR to reflect the revised licensing basis authorized by this amendment in accordance with 10 CFR 50.71(e).

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: September 5, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 167

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
VIII	VIII
----------	3/4 7-9a
B 3/4 7-3	B 3/4 7-3
----------	B 3/4 7-3a
----------	B 3/4 7-3b

SECTION		PAGE
3/4.7 PLANT SYSTEMS (Continued)		
	ACTIVITY	3/4 7-7
	MAIN STEAM LINE ISOLATION VALVES	3/4 7-9
	MAIN FEEDWATER ISOLATION VALVES	3/4 7-9a
3/4.7.2	STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION	3/4 7-10
3/4.7.3	COMPONENT COOLING WATER AND AUXILIARY COMPONENT COOLING WATER SYSTEMS	3/4 7-11
3/4.7.4	ULTIMATE HEAT SINK	3/4 7-12
3/4.7.5	FLOOD PROTECTION	3/4 7-15
3/4.7.6	CONTROL ROOM AIR CONDITIONING SYSTEM	3/4 7-16
3/4.7.7	CONTROLLED VENTILATION AREA SYSTEM	3/4 7-19
3/4.7.8	SNUBBERS	3/4 7-21
3/4.7.9	SEALED SOURCE CONTAMINATION	3/4 7-27
3/4.7.12	ESSENTIAL SERVICES CHILLED WATER SYSTEM	3/4 7-43
3/4.8 ELECTRICAL POWER SYSTEMS		
3/4.8.1	A.C. SOURCES	
	OPERATING	3/4 8-1
	SHUTDOWN	3/4 8-8
3/4.8.2	D.C. SOURCES	
	OPERATING	3/4 8-9
	SHUTDOWN	3/4 8-12

MAIN FEEDWATER ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.6 Each Main Feedwater Isolation Valve (MFIV) shall be OPERABLE.

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

ACTION:

Note: Separate Condition entry is allowed for each valve.

With one or more MFIV inoperable, close and deactivate, or isolate the inoperable valve within 72 hours and verify inoperable valve closed and deactivated or isolated once every 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The provisions of Specification 3.0.4 do not apply.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each main feedwater isolation valve shall be demonstrated OPERABLE:

a. By verifying isolation $\leq$ 5.0 seconds when tested pursuant to Specification 4.0.5.

b. By verifying actuation to the isolation position on an actual or simulated actuation signal at least once per 18 months.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss-of-offsite electricalpower. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVE

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

3/4.7.1.6 MAIN FEEDWATER ISOLATION VALVES

The Main Feedwater Isolation Valves (MFIVs) isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). Closure of the MFIVs terminates flow to both steam generators, mitigating the consequences for feedwater line breaks (FWLBs). Closure of the MFIVs effectively terminates the addition of main feedwater to an affected steam generator, limiting the mass and energy release for Main Steam Line Breaks (MSLBs) or FWLBs inside containment, and reducing the cooldown effects for MSLBs.

The MFIVs isolate the non-safety related feedwater supply from the safety related portion of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of Emergency Feedwater (EFW) to the intact steam generator.

One MFIV is located on each MFW line, outside, but close to, containment. The MFIVs are located upstream of the EFW injection point so that EFW may be supplied to a steam generator following MFIV closure. The piping volume from the valve to the steam generator must be accounted for in calculating mass and energy releases, and refilled prior to EFW reaching the steam generator following either a MSLB or FWLB.

The MFIVs close on receipt of a Main Steam Isolation Signal (MSIS) generated by either low steam generator pressure or high containment pressure. The MFIVs may also be actuated manually from the control room. The MSIS also actuates the Main Steam Isolation Valves (MSIVs), Main Feedwater Regulating Valves (MFRVs) and Startup Feedwater

3/4.7.1.6 MAIN FEEDWATER ISOLATION VALVES (con't)

Regulating Valves (SFRVs) to close. The Feedwater Regulating Valve Bypass Valves are normally closed and deactivated during power operation.

In MODES 1, 2, 3, and 4, the MFIVs are required to be OPERABLE, except when they are closed and deactivated or isolated by either a closed manual valve or closed and deactivated automatic valve, in order to limit the amount of available fluid that could be added to the Steam Generator and/or containment in the case of a secondary system pipe break inside containment. When a MFIV is closed and deactivated or isolated by a closed manual valve or closed and deactivated automatic valve, it is already performing its safety function.

In MODES 5 and 6, residual heat removal is through the Shutdown Cooling System and MFW is not required. Therefore, the MFIVs are normally closed.

With one MFIV inoperable, action must be taken to close or isolate the inoperable valve within 72 hours. When the valve is closed or isolated, it is performing the required safety function (e.g., to isolate the main feedwater line) and continued operation in the applicable MODES is allowed.

The 72 hour Completion Time takes into account the back up capability afforded by the OPERABLE MFRVs and the SFRVs, diversity of actuation signals, and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable to return the MFIV to OPERABLE status, close the MFIV, or otherwise isolate the affected flow path.

Inoperable MFIVs that cannot be restored to OPERABLE status within 72 hours, but are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day time is reasonable in view of valve status indications available in the control room, and other administrative controls to ensure that these valves are closed or isolated.

If the MFIVs cannot be restored to OPERABLE status, closed, or isolated in the required time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 in the following 30 hours. The allowed completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

The TS is annotated with a 3.0.4 exemption, allowing entry into the applicable MODES to be made with an inoperable MFIV closed or isolated as required by the ACTIONS. The ACTIONS allow separate condition entry for each valve by using "With one or more MFIV...". This prevents immediate entry into TS 3.0.3 if both MFIVs are declared inoperable.

3/4.7.1.6 MAIN FEEDWATER ISOLATION VALVES (con't)

The Surveillance Requirement to verify isolation in less than or equal to 5 seconds is based on the time assumed in the accident and containment analyses. The static test demonstrates the ability of the MFIVs to close in less than or equal to 5 seconds under design basis accident conditions. The MFIVs should not be tested at power since even a partial stroke exercise increases the risk of a valve closure with the plant generating power and would create added cyclic stresses. The Surveillance to verify each MFIV can close on an actual or simulated actuation signal is normally performed when the plant is returning to operation following a refueling outage. Verification of valve closure on an actuation signal is not required until entry into Mode 3 consistent with TS 3.3.2. The 18 month frequency is based on the refueling cycle. Verification of closure time is performed per TS 4.0.5. This frequency is acceptable from a reliability standpoint and is in accordance with the Inservice Testing Program.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator secondary pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitation to 115°F and 210 psig is based on a steam generator RTNDT of 40°F and is sufficient to prevent brittle fracture. Below this temperature of 115°F the system pressure must be limited to a maximum of 20% of the secondary hydrostatic test pressure of 1375 psia (corrected for instrument error). Should steam generator temperature drop below 115°F an engineering evaluation of the effects of the overpressurization is required. However, to reduce the potential for brittle failure the steam generator temperature may be increased to a limit of 200°F while performing the evaluation. The limitations on the primary side of the steam generator are bounded by the restrictions on the reactor coolant system in Specification 3.4.8.1.

3/4.7.3 COMPONENT COOLING WATER AND AUXILIARY COMPONENT COOLING WATER SYSTEMS

The OPERABILITY of the component cooling water system and its corresponding auxiliary component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the safety analyses.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 167 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 15, 1999, as supplemented by letter dated March 29, 2000, Entergy Operations, Inc. (the licensee or Entergy), submitted a request for changes to the Waterford Steam Electric Station, Unit 3 (Waterford 3), Technical Specifications (TSs). The requested changes would add a new TS for the Main Feedwater Isolation Valves (MFIVs) consistent with the guidelines of TS 3.7.3, Main Feedwater Isolation Valves, in NUREG 1432, "Standard Technical Specifications - Combustion Engineering Plants." The licensee also proposed to take credit for two non-safety related features to reduce the pressure differential across the MFIV disc during accident conditions and thereby increase the MFIV closure speed to support the safety analysis closure time of five seconds. These non-safety related features are: (1) reactor trip override (RTO), and (2) auxiliary feedwater (AFW) pump high pressure trip. An associated TS BASES section was also added to the Waterford 3 TS.

2.0 BACKGROUND

The MFIVs were originally designed for closure against a differential pressure of 1,400 pounds per square inch, gauge (psig), assuming a 0.2 valve friction coefficient. During a Nuclear Regulatory Commission (NRC) staff inspection in 1997, the licensee discovered that if only safety-related system functions were credited, the differential pressure across the MFIV could be postulated to reach values higher than 1,400 psig. Furthermore, based on industry and Electric Power Research Institute testing, the recommended bounding friction coefficient for this type and size of valve is 0.4. Because of this design deficiency, the MFIV may not be able to meet its design basis five-second closure time under certain design basis accident conditions, if only safety related functions are credited. This could potentially allow greater mass and energy into the containment or increase primary system cooldown, during main steam line break (MSLB) and feedwater line break (FWLB) events, than is currently analyzed.

The licensee proposes to take credit for RTO to reduce the speed of the steam generator feed pumps (SGFP) during a design basis accident at full power, and the AFW pump high discharge pressure trip to stop the AFW pump during an accident at low power. The speed reduction and

the trip function will decrease the pressure developed across the MFIV disc and facilitate more rapid valve closure, thereby avoiding costly valve/actuator replacement. The purpose of the RTO is to prevent overcooling of the reactor coolant system after a reactor trip. This is accomplished by limiting the feedwater flow to a minimum rate so that it will slowly refill the steam generators and make up for decay heat.

3.0 EVALUATION

3.1 Deterministic Evaluation

NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976, Memorandum from Director, NRR to NRR Staff," Issue Number 1, "Treatment of Non-Safety Grade Equipment in Evaluations of Postulated Steam Line Break Accidents," and NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Plants," Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)," item II.i, state:

> For postulated instantaneous pipe failures in seismically qualified portions of the main steam line (inside containment and upstream of the MSIVs), only safety grade equipment should be assumed operative. If, in addition, a single malfunction or failure of an active component is postulated, credit may be taken for the use of a backup non safety grade component to mitigate the consequences of the break.

The licensee has taken credit for the main feedwater regulating valves (MFRV) and startup feedwater regulating valves (SFRV) as a backup to the MFIV to meet the single failure criterion, as stated in NUREG-0138 and SRP Section 15.1.5. The MFRV and SFRV are non-safety related equipment. This TS change will credit two additional pieces of non-safety grade instrumentation features (RTO and AFW pump high pressure trip) in the accident evaluation to meet the design basis five-second closure time for the MFIV. The MFIV, MFRV and SFRV are designed to close within five seconds when they receive a main steam isolation signal (MSIS). The MFRV and SFRV are furnished with emergency closure circuits so that the closure of these valves is actuated through override of their normal control signal.

The licensee postulated four scenarios with the two non-safety related features in order to estimate the required thrust, available thrust with actual system pressures, and containment peak pressure. These four scenarios are:

(1) MSLB inside containment - 100 percent Power SGFP operational
(2) FWLB inside containment - 100 percent Power SGFP operational
(3) MSLB inside containment - 0 percent Power AFW pump operational and
(4) FWLB inside containment - 0 percent Power AFW pump operational.

The results are as follows:

Scenario	Required Thrust (lbs)	Available Thrust (lbs) 2 Accumulators	RTO / AFW Pump Trip Credit?	Closure Time
MSLB (SGFP)	100,398	100,086	No	5 seconds
FWLB (SGFP)	98,478	100,086	Yes	5 seconds
MSLB (AFW)	80,945	100,086	Yes	5 seconds
FWLB (AFW)	106,804	100,526	Yes	30 seconds

Where
lbs = pounds

For the MSLB at 100 percent power, the MFIV will close within five seconds without credit for RTO due to the lower pressure differential pressure across the valve from the residual pressure existing in the ruptured steam generator. RTO will be credited for MFIV closure in all other scenarios and the resulting required thrust is less than the available thrust with two exceptions. The required thrust is slightly greater than the available thrust for the MSLB during SGFP operation, but is considered acceptable due to the use of a conservatively bounding friction coefficient, calculational methods, nitrogen starting pressures, feedwater temperatures, and valve packing assumptions. For the FWLB with AFW pump in operation, MFIV closure with credit for RTO/AFW pump trip will take 30 seconds. The MFIV will experience full differential pressure in this scenario, and will have a longer closure time and continued flow of AFW into the containment. However, the water being pumped into the containment is below the saturation temperature of the containment atmosphere. Therefore, this flow will condense the steam and would not contribute to increased containment pressure. The peak containment pressure is bounded by the MSLB for this plant. The peak pressure for the FWLB is several psi below that for the MSLB.

The MSIV, MFIV, MFRV, and SFRV are designed to receive the same closure signal, MSIS, and they all close within five seconds. The MFRVs do not depend upon RTO and the AFW pump pressure trip instrumentation for their closure time. The MSIS components undergo response time testing, channel functional tests, and relay functional tests every 18 months, per the Waterford 3 TS. In addition, testing of the MFIV, MFRV, and SFRV will be performed in accordance with the inservice testing (IST) program. These TS and IST requirements provide sufficient assurance that, if the MFIV failed to close, the other valves would provide proper isolation of the feedwater system after a reactor trip to prevent over cooling of the reactor coolant system. The RTO and AFW pump high pressure trip are also subjected to a testing program similar to comparable safety related instrumentation to ensure their functionality. The operating history of the RTO and AFW pump trip instrumentation shows high reliability and, thereby, provides assurance that they will meet their functional requirements. Therefore, the staff finds that the non-safety grade instrumentation components in the steam and feedwater systems will function under the required accident conditions for which they are called upon, and are acceptable.

3.2 Probabilistic Safety Assessment Evaluation

The staff used a three-tiered approach to evaluate the risk associated with the proposed TS changes. The first tier evaluated the Probabilistic Safety Assessment (PSA) model and the impact of the allowed outage time (AOT) implied extension (four hours is now informally used) to 72 hours for the MFIV on plant operational risk. The evaluation of the PSA model relied, in part, on cross-comparison approaches with similar plants. The second tier addressed the need to preclude potentially high risk configurations, and is directed toward the need to provide additional constraints or compensatory actions that, if implemented, would avoid or reduce the probability of a risk-significant configuration during the time when a MFIV is out of service.

3.2.1 Tier 1 Evaluation and PSA Quality

Failure to close the MFIVs is not included in the Level 1 PSA model for core damage frequency (CDF). The only area of the model in which failure to close the MFIVs is modeled is in the containment isolation fault tree, which is used in calculating Large Early Release Frequency (LERF). Therefore, LERF is the only PSA result that would be impacted if a MFIV failed to close.

The failure of a MFIV to close in five seconds is not in the Level 1 model because it has no effect on core damage. The MFIV will eventually close when the SGFP speed reduces due to a loss of steam after the MSIV closes. This is expected to occur within approximately one minute. The short additional time that the MFIV may remain open has no impact on core damage. The only postulated way core damage could occur from a MSLB or FWLB event is if failure of emergency feedwater (EFW) were to occur, eliminating decay heat removal. The change under consideration does not affect the ability to inject water to the steam generators and does not affect EFW. Additionally, EFW is not normally injected into the affected steam generator. If operators were for some reason to decide to feed the affected steam generator, and the MFIV had not yet closed, the check valve upstream of the MFIV would prevent EFW diversion into the main feedwater system. In conclusion, failure of the MFIV to close within five seconds on demand does not affect the plant's CDF.

Because there is no impact on CDF, the impact on LERF should also be small. Since the failure of the MFIV to close is located in the containment isolation fault tree, the licensee has quantified the change in LERF. Because the failure of the MFIV to close does not affect containment bypass or high-pressure-melt-ejection sequences and the CDF has not been impacted, the contribution from the breach of containment isolation standpoint is the only factor to impact the LERF. The licensee's calculated change in the top gate failure probability was 2.2E-5. Multiplying this by the current Waterford 3 baseline CDF of 2.54E-5/reactor year (RY), yields a change in LERF of 5.59E-10/RY, which is well within the "very small" guideline value of Regulatory Guide (RG) 1.174 (1.0E-7/RY). Therefore, the staff judges that since the change in LERF is so small, the incremental conditional large early release probability is also small and within the 5E-8 guideline of RG 1.177.

Three levels of review were performed on the original Waterford 3 Individual Plant Examination (IPE) model. 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 the CAFTA/DOS software. Waterford 3 plant personnel not involved in the PSA development performed the second level of review.

This review group consisted of individuals from the Waterford 3 Operations, Licensing, Engineering, and Training departments, 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 review of 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/CQUANT32 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 in the Revision 2, Change 1 stage. Some of the major changes that have been incorporated since the IPE submittal are as follows: (1) the elimination of asymmetries across multiple train systems (allowing the swing trains to recover either A or B trains, rather than only one), (2) the inclusion of additional direct current power dependencies on applicable systems, (3) the incorporation of a detailed convolution methodology for calculating offsite power recovery factors, and (4) the updating 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., misclassification of a valve as a motor operated valve instead of an air operated valve, or basic event description changes).

Every change to the PSA model since the IPE has been prepared by a Waterford 3 PSA engineer; reviewed by a separate, independent PSA engineer; and approved by the Manager, Safety and Engineering Analysis. Cross comparisons of Waterford 3 risk-related results that support AOT extensions with other Combustion Engineering (CE) Owners Group (CEOG) plants have been made. This provides 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, involving mainly non-modeling of air operated valve common cause failure. The licensee has determined that this is not a meaningful contributor to incremental conditional core damage probability (ICCDP) or incremental conditional large early release probability (ICLERP). The team also identified some conservatisms. Entergy will develop a plan to prioritize the PSA Certification Team's concerns and implement the necessary improvements. Assurance that changes to the as-built, 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 Configuration Risk Management Program (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 null change in CDF, and therefore in ICCDP, resulting from the proposed TS change coupled with the extremely small change in LERF, and therefore in ICLERP estimated for an AOT of 72 hours, to be reasonable. Based on the above discussion, the staff finds the PSA model used by the Waterford 3 licensee to be acceptable and concludes that there is minimal impact from the 72 hour completion time for the MFIV on plant operational risk.

3.2.2 Tier 2 Evaluation

The licensee did not identify any risk-significant configurations associated with the proposed 72 hour MFIV AOT. 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-informed configuration. The staff concurs with these findings.

3.2.3 Tier 3 Evaluation

The licensee proposes to implement a CRMP and to establish the CRMP requirements in Site Directive W2.502. The purpose of the CRMP is to ensure that a proceduralized PSA-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 PSA, in addition to those SSCs considered of high safety significance per RG 1.160, Revision 2 (the Maintenance Rule Regulatory Guide), that are not modeled in the PSA.

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 PSA-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 pre-planned 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.

The staff finds the CRMP to be consistent with staff guidance in RG 1.160 and, therefore, acceptable.

4.0 SUMMARY

The staff has evaluated the Waterford 3 licensee's proposed addition of TS LCO 3.7.1.6 for the MFIVs with an AOT of 72 hours for compliance with regulatory requirements as documented in this evaluation and concludes that it is acceptable. This determination is based on the following:

1. The staff finds acceptable the PRA model used by the Waterford 3 licensee and also concludes that there is minimal impact from the AOT for the MFIV on plant operational risk (Tier 1 evaluation).

2. 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).

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

4. The use of the RTO/AFW pump high discharge pressure trip is consistent with staff guidance and these features are sufficiently reliable to meet their intended design and performance requirements of a MFIV closure within five seconds for postulated MSLB and MFLB accidents.

By approving this amendment, the staff is not removing the licensing basis requirement that only safety related equipment be used to evaluate the plant response to design basis accidents or changing the valve response requirement of closure within five seconds.

A revision to the Final Safety Analysis Report will be made to address the changes associated with this TS Change Request and the resolution of the unreviewed safety question.

5.0 STATE CONSULTATION

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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The 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 4275, dated January 26, 2000). The March 29, 2000, supplement provided clarifying information that did not expand the scope of the original Federal Register notice, or change the scope of the initial proposed no significant hazards consideration determination. 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.

7.0 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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