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

July 6, 2000

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT RE: REDUCTION IN OPERABLE CONTAINMENT FAN COOLERS IN THE CONTAINMENT COOLING SYSTEM (TAC NO. MA6997)

Dear Mr. Dugger:

The Commission has issued the enclosed Amendment No. 165 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3 (Waterford 3). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 18, 1999, as supplemented by letters dated May 16, 2000, and June 1, 2000.

The amendment modifies Technical Specification (TS) 3.6.2.2 Limiting Condition for Operation to allow Waterford 3 to operate with two independent trains of containment cooling, consisting of one cooler per train, operable during Modes 1, 2, 3, and 4. Associated changes to the TS Bases have been incorporated.

The calculations performed, in which measurement uncertainties were included, show that containment pressure and temperature limits are satisfied for both the loss of coolant accident and main steam line break accident, and these calculations are the basis for the staff's approval of this specific license amendment. The staff has not, however, made a final determination of the acceptability of this approach on a generic basis. Waterford 3 should discuss with the staff the appropriate forum for pursuing the use of graded uncertainties.

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. 165 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

July 6, 2000

Mr. Charles M. Dugger
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17265 River Road
Killona, LA 70066-0751

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Sincerely,

N. Kalyanam, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
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Docket No. 50-382

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2. Safety Evaluation

cc w/encls: See next page

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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. 165
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 October 18, 1999, as supplemented by letters dated May 16, 2000, and June 1, 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. 165 , 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

Mohan C. Thadani, Acting 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: July 6, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 165

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 6-18	3/4 6-18
B 3/4 6-4	B 3/4 6-4
B 3/4 6-5	B 3/4 6-5

CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Two independent trains of containment cooling shall be OPERABLE with one fan cooler to each train.

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

ACTION:

With one train of containment cooling inoperable, restore the inoperable train to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable containment cooling train to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each train of containment cooling shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Starting each operational fan not already running from the control room and verifying that each operational fan operates for at least 15 minutes.

   2. Verifying a cooling water flow rate of greater than or equal to 625 gpm to each cooler.

b. At least once per 18 months by:

   1. Verifying that each fan starts automatically on an SIAS test signal.

   2. Verifying a cooling water flow rate of greater than or equal to 1200 gpm to each cooler.

   3. Verifying that each cooling water control valve actuates to its full open position on a SIAS test signal.

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 $\leq$ 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.

The Containment Cooling System consists of two redundant trains and is designed such that a single failure does not degrade the systems' ability to provide the required heat removal 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). An operable train of containment cooling consists of one of the two fans and its associated cooler. One Containment Cooling train, consisting of one fan and its associated cooler, and a Containment Spray train has sufficient capacity to meet post accident heat removal requirements and maintain containment temperatures and pressures below the design values.

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

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

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.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 165 TO

FACILITY OPERATING LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By letter dated October 18, 1999, as supplemented by letters dated May 16 and June 1, 2000, Entergy Operations, Inc. (EOI or the licensee) proposed a change to the Waterford Steam Electric Station, Unit 3 (Waterford 3), Technical Specification (TS) 3.6.2.2, that would allow operation of Waterford 3 with two independent trains of containment fan coolers (CFC), with each train required to have one operable fan cooler. The current TS require two fan coolers in each train to be operable.

By letter dated April 27, 2000, the licensee requested that the Nuclear Regulatory Commission (NRC) exercise discretion not to enforce compliance with the actions required in TS 3.6.2.2. This request was based on the failure of CFC "C" on April 24, 2000. The letter documented information previously discussed with the NRC in a conference call on April 26, 2000. The NRC granted the Notice of Enforcement Discretion (NOED) verbally on April 27, 2000, and confirmed by letter dated May, 1, 2000, the granting of the NOED. The NOED provides NRC discretion not to enforce compliance with the required actions of TS 3.6.2.2 which require shutdown of Waterford 3 with the CFC "C" inoperable more than 72 hours. This NOED is to be effective for the period until Waterford 3 TS Change Request NPF-38-224 (the subject of this safety evaluation report input) is approved or an outage of sufficient length occurs to accommodate repair of the CFC "C". In the meantime, Waterford 3 will continue to operate with the requirement that one operable CFC per train must be operable.

Responses to staff requests for additional information (RAI) were provided by the licensee by letters dated May 16, 2000, and June 1, 2000.

The staff evaluation of the licensee's proposed change to the Waterford 3 TS is provided below.

2.0 BACKGROUND

Along with the Containment Spray System, the Containment Cooling System (CCS) is part of the Containment Heat Removal System (CHRS) at Waterford 3. The safety-related function of the CHRS is to remove heat from the containment atmosphere during a design basis accident in order to maintain the containment pressure and temperature within acceptable limits. The CCS in turn

consists of four CFCs, two per train. Each train is powered by an independent safety-related bus. The CCS starts the CFCs upon receipt of a safety injection actuation signal.

TS 3.6.2.2 presently requires that two independent trains of the CCS be operable with two fan coolers in each train. With one train of CCS inoperable, the required action of TS 3.6.2.2 is to restore the inoperable train to operable status within 72 hours or be in at least MODE 3 within the next six hours. If the inoperable train cannot be restored to operable status within the next 48 hours, the plant must be placed in cold shutdown within the next 30 hours.

The licensee is proposing to revise TS 3.6.2.2 so that two independent trains shall be operable with one fan cooler in each train.

In order to justify this change, the licensee performed calculations to show that, with one fan cooler per train, for the design basis loss-of-coolant accident (LOCA) and the design basis main steam line break (MSLB), the containment will be maintained below its design basis temperature and pressure. Typically, these calculations are done by calculating the mass and energy of the reactor coolant (for the LOCA) or steam (for the MSLB) which is discharged into the containment and then calculating the increase in containment pressure and temperature due to this added mass and energy.

In order to demonstrate that the containment pressure and temperature remain below design limits, the licensee performed calculations with NRC-approved ABB/CE computer codes for the mass and energy release calculations, and used the GOTHIC computer code to analyze the containment response.

The licensee included a new method of treating measurement uncertainties in these calculations. For the reasons discussed below, the staff is not approving this method in this safety evaluation report input. However, the licensee also performed the calculations treating measurement uncertainties consistent with the current Waterford 3 licensing basis and they are the basis for the staff approval of the licensee's request.

The licensee's mass and energy discharge calculations for the MSLB used a main steam isolation valve (MSIV) closure time of eight seconds (seven second MSIV closure time plus one second for signal delay) rather than the current value of five second closure time (four second MSIV closure time plus one second for signal delay).

3.0 EVALUATION

The licensee stated that the major differences between the Final Safety Analysis Report (FSAR) analyses and the analyses of the October 18, 1999, submittal for the LOCA and the MSLB are: (1) the FSAR analyses assume two CFCs per train are operable while the new analyses assume one fan cooler per train is operable, (2) there are new calculated mass and energy release values, and (3) different values used for the parameters in Table 1 of the June 1, 2000, licensee response to Question 6 of the staff's RAI, reproduced below, include a different method of treating instrumentation uncertainties. The first two differences are discussed in Sections 3.1 and 3.2, respectively, while the treatment of instrument uncertainties is addressed in Section 3.3.

Question 6:

Provide a summary describing the key parameter uncertainties and the impact of the uncertainties on the limiting containment pressure and temperature response. Also provide peak containment pressure results for suction leg and discharge leg breaks.

Response:

The following table provides the values and their associated instrument uncertainties for key parameters used in the supplemental GOTHIC analysis to assess the impact of uncertainties. The uncertainty values, shown in Table 1 and used in this GOTHIC analysis, for initial containment temperature and CS [Containment Spray] riser level are different than the values used in the previous FSAR analyses and reflected in Table 5 of the TSCR [TS Change Request] submittal. In the FSAR analyses, an instrument uncertainty of 6 °F, corresponding to the average of three loops, was used for the initial containment temperature. Note that containment temperature at Waterford 3 is currently measured by averaging the values from three loops. The temperature uncertainty used in the GOTHIC analyses corresponds to the calculated value for an individual loop. In the FSAR analyses, a conservative CS riser level instrument uncertainty of 10 ft [feet] was used. The calculated CS riser level uncertainty is 7 ft. In addition, uncertainty for the initial containment pressure was not considered in the FSAR analyses. However, it was added to the supplemental GOTHIC uncertainty analysis.

Table 1

Parameter	w/o Uncertainty	Uncertainty	With Uncertainty
Initial Containment Temperature	120 °F	+7.5 °F	127.5 °F
CCW Temperature	115 °F	+5 °F	120 °F
CCW Flow	1200 gpm	-100 gpm	1100 gpm
Initial CS Riser Level	149.5 ft	-7 ft	142.5 ft
Initial Containment Pressure	15.7 psia	+0.11 psi	15.81 psia

Where:
CCW = Component Cooling Water
gpm = gallons per minute
psia = pounds per square inch, absolute
psi = pounds per square inch

3.1 LOCA

The initial mass and energy release during the blowdown phase of the LOCA is calculated with the CEFLASH 4A computer code. This code is used for emergency core cooling system (ECCS) performance licensing calculations. In order to conservatively bias the containment mass and energy discharge calculations, the licensee stated that several changes to the modeling were made to maximize the heat transfer from the core to the coolant during LOCA. The licensee's May 16, 2000, response to Question 1 of NRC's May 12, 2000, RAI provided additional details. The CE/ABB FLOOD3 and CONTRANS codes were used to calculate the mass and energy release to the containment after the blowdown phase. The licensee's October 18, 1999, letter provides references to previous staff approval of these codes. The licensee used the ANS 5.1-1979 decay heat model with a 2σ uncertainty rather than the decay heat model previously included in the CONTRANS code. This decay heat model is widely used by the industry for licensing calculations and is acceptable.

The GOTHIC code was developed for the Electric Power Research Institute and is maintained under a 10 CFR Part 50 Appendix B Quality Assurance Program by its developer, Numerical Applications, Inc. It has been extensively verified against analytical solutions and experiments. Although the NRC has not specifically approved GOTHIC, the NRC has approved its application to problems similar to those analyzed in this submittal. The licensee has used the GOTHIC code to calculate the containment response for both the LOCA and MSLB accidents.

The licensee's May 16, 2000, letter to the NRC provides comparisons between GOTHIC and CONTEMPT LOCA and MSLB calculations for Waterford 3. These results show good agreement in both magnitude and trend. The CONTEMPT computer code is the licensee's original and current licensing basis code. It was developed by the NRC and is listed in their Standard Review Plan Section as an approved code.

The licensee performed all LOCA calculations at a power level 8% greater than the current licensed power level in preparation for a power uprate request.

All mass and energy calculations for the LOCA cases were performed with both maximum and minimum safety injection (SI) flow assumptions. The containment pressure and temperature analyses for the LOCA cases were analyzed assuming a loss of offsite power and a failure of one train of the containment heat removal system (leaving one containment spray train and one containment fan cooler operable) for both the maximum and the minimum SI cases. The licensee points out that this is conservative for the maximum SI flow cases since the maximum SI flow assumption precludes a single failure of a power bus that would cause a failure of both a containment spray train and a containment cooling system train since both emergency diesel generators are assumed operable for this case.

The licensee calculated the mass and energy release into containment for a double ended discharge leg slot break (DEDLSB), a double ended suction leg slot break, and a double ended hot leg slot break (DEHLSB). In response to a staff question, the licensee performed "an informal" GOTHIC analysis using mass and energy values provided in the June 1, 2000, submittal which shows that for the discharge and suction leg break locations, the slot break produces a peak containment pressure greater than 0.5 psi above that for the guillotine break. However, for the hot leg break, the slot break pressure is lower than the guillotine by less than 0.05 psi. The staff agrees with the licensee that this is within the accuracy (and conservatism) of the analyses.

However, it does highlight the need to consider a spectrum of break sizes and locations in performing calculations of this type.

The licensee calculated values of the LOCA peak containment pressure and found the LOCA peak containment temperatures are less than the design limits of 44 psia and 413.5 °F, respectively. The latter is the environmental qualification design temperature, not the containment design temperature limit. The table below, taken in part from the licensee's June 1, 2000, response to the staff's May 23, 2000, RAI, summarizes the licensee's LOCA results.

Table 2

Parameter	w/o Uncertainty	w/Uncertainty	Difference
Limiting LOCA Peak Pressure (DEHLSB)	35.20 psig	35.34 psig	0.14 psi
Limiting LOCA P @ 24 hours (DEDLSB, min SI)	15.50 psig	15.90 psig	0.40 psi

Where P=Pressure

Note that the licensee has not proposed changing the TS value of the containment peak pressure for containment leakage rate testing, P_a. Therefore, the value of P_a will remain at a pressure higher than the licensee's new value.

The licensee's calculations showed a large difference in both magnitude and timing of the peak containment pressure between the new calculations and the FSAR analyses (see Table 1 of the licensee's October 18, 1999, submittal). In addition, the location of the break, which gives the peak containment pressure, has changed (hot leg break in the October 18, 1999, submittal versus suction leg break in the FSAR). The licensee states that the difference between the FSAR values and the new results is primarily due to the new mass and energy data which calculate a lower mass and energy into the containment during the reflood period. The worst case FSAR analysis and the worst case analysis in the October 18, 1999, submittal for peak containment pressure again are for different break locations (Double Ended Suction Leg Slot Break and Double Ended Hot Leg Slot Break, respectively).

For the calculation of peak containment pressure at 24 hours after accident initiation, the licensee assumed that the operable fan cooler was the "B" fan cooler which, because it is partially flooded during the LOCA, has less area available for cooling. The containment pressure at 24 hours must be reduced to less than half the containment peak pressure or 16.64 psig. (This value does not include measurement uncertainties.) The calculated containment pressure at 24 hours is less than this value. Therefore, it is conservative to assume half the peak pressure for offsite and control room dose calculations.

3.2 MAIN STEAM LINE BREAK

The mass and energy release calculations for the MSLB event were calculated using the SGNIII computer code. This is an NRC-approved computer code which is the current Waterford 3 licensing basis code for secondary pipe break analysis. The licensee's submittal references the NRC Safety Evaluation Report approving SGNIII. The containment calculations have been performed with the GOTHIC code.

The MSLB analyses were done at 102%, 75%, and 50% of the current licensed power level. The zero power case is sometimes limiting for MSLBs. However, the licensee states that the results of the new analyses demonstrate that the higher power levels result in larger values of peak containment pressure. This is consistent with the FSAR results.

The worst single failure is loss of one containment heat removal train leaving one containment spray pump and one CFC operable. The analyses assumed superheated steam with no entrainment. This is conservative. The analyses also did not contain an allowance for re-evaporation of the condensate (8%) from the passive heat sinks of the containment which is permitted according to the guidelines of NUREG-0588, Appendix B. This is also conservative.

The licensee's mass and energy discharge calculations for the MSLB used an MSIV closure time of eight seconds (seven second MSIV closure time plus one second for signal delay) rather than the current value of five second closure time (four second MSIV closure time plus one second for signal delay). A longer closing time for the MSIV leads to a longer blowdown from the intact steam generator and thus, a higher pressure in containment. This change is acceptable since the licensee's calculations show that the pressure and temperature in containment remain below the design pressure and temperature following a MSLB with the longer MSIV closing time. The change in stroke time does not involve a change to the TS since TS Surveillance Requirement 4.6.3.3, which requires the isolation time of power-operated or automatic containment isolation valves to be determined, does not specify the timing for these valves.

The table below, which is taken in part from the licensee's June 1, 2000, response to staff RAI, provides the results of the licensee's MSLB calculations.

Table 2

Parameter	w/o Uncertainty	w/Uncertainty	Difference
MSLB Peak Pressure	42.68 psig	43.26 psig	0.58 psi
MSLB Peak Temperature	397.4 °F	402.60 °F	5.2 °F

3.3 INSTRUMENTATION UNCERTAINTIES

The LOCA and MSLB analyses do not explicitly include measurement uncertainties for five input parameters: initial containment temperature, CCW flow, CCW temperature, initial containment pressure, and initial containment spray riser level. The licensee's submittal, dated October 18, 1999, states that this is reasonable because of the large margin available between

the design pressure and the actual failure pressure of containment. The licensee refers to this method of treating measurement uncertainties as a graded approach. The graded approach was the subject of a December 2, 1999, meeting between the licensee and the staff at NRC headquarters and a February 22, 2000, meeting between the licensee and the staff at NRC Region IV. This topic was also the subject of an NRC inspection, which took place February 28 through March 3, 2000, (March 30, 2000, letter from Dr. Dale A. Powers, USNRC, to Charles M. Dugger, Vice president Operations - Waterford 3).

The method proposed by the licensee, as described in the October 18, 1999, submittal and later correspondence, is to not explicitly account for measurement uncertainties for the five items listed above and included in Table 1 of the licensee's June 1, 2000, response to staff RAI (reproduced above) since the impact of these uncertainties is small, as can be seen from Table 2 of the licensee's June 1, 2000, response to staff RAI (reproduced above). In terms of containment pressure for a MSLB, the difference between an analysis including uncertainties and an analysis without these uncertainties is 0.58 psi. The licensee states that this is much less than the difference between the containment design pressure and the actual failure pressure of the containment and that, therefore, the measurement uncertainties of the five parameters listed above may be neglected. However, this is combining a design basis analysis with a risk-based success criterion.

Because of the short review time available since this application became the basis for the licensee's NOED request, the staff has not made a final determination of the acceptability of this approach. The calculations performed by the licensee, in which measurement uncertainties were included, show that containment pressure and temperature limits are satisfied for both the LOCA and MSLB accident, and these calculations are the basis for the staff's approval of this license amendment.

The licensee should discuss with the staff the appropriate forum for pursuing the use of graded uncertainties.

4.0 SUMMARY

The licensee's submittal, dated October 18, 1999, states that even though the LOCA and MSLB analyses do not explicitly include measurement uncertainties for five input parameters detailed in Section 3.3 above, this is acceptable because of the large margin available between the design pressure and the actual failure pressure of containment. The licensee's letter dated June 1, 2000, states the calculations performed in which measurement uncertainties were included, show that containment pressure and temperature limits are satisfied for both the LOCA and MSLB accident and that the impact of these uncertainties is small. These calculations are the basis for the staff's approval of this license amendment. For future use, the licensee should discuss with the staff the appropriate forum for pursuing the use of graded uncertainties.

Based on the above, the licensee has provided analyses which demonstrate that Waterford 3 can operate safely with one operable CFC in each train of the CCS. The calculations which justify this operation were performed with acceptable methods and with an acceptable level of conservatism.

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 6407, dated February 9, 2000). The May 16 and June 1, 2000, supplements did not expand the scope of the application as noticed or change the 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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