March 29, 2005

Mr. Joseph E. Venable
Vice President Operations
Entergy Operations, Inc.
17265 River Road
Killona, LA 70066-0751

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 (WATERFORD 3) - ISSUANCE OF AMENDMENT RE: FULL-SCOPE IMPLEMENTATION OF AN ALTERNATIVE ACCIDENT SOURCE TERM (TAC NO. MC3789)

Dear Mr. Venable:

The U. S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 198 to Facility Operating License (FOL) No. NPF-38 for the Waterford 3. This amendment consists of changes in response to your application dated July 15, 2004, and supplemented by letters dated August 19, September 1, September 14, October 13, and October 19, 2004.

Entergy Operations, Inc., (Entergy or the licensee) had by application dated November 13, 2003, requested an extended power uprate (EPU) of 8 percent at Waterford 3. Consistent with the existing Waterford 3 licensing basis, the EPU presented doses to control room personnel for only the large break loss-of-coolant accident (LOCA) and fuel handling accident (FHA). Entergy stated that control room doses for these and other accidents would be addressed as necessary in conjunction with the response to Generic Letter (GL) 2003-01, "Control Room Habitability."

In response to GL 2003-01, Entergy performed control room tracer gas testing in April 2004. Based on the results of the test, Entergy decided to implement an alternative source term (AST) in the calculation of accident doses to control room personnel. Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Paragraph 50.90, Entergy requested, by application dated July 15, 2004, approval of an amendment for Waterford 3, indicating the intention to implement an AST, as permitted by 10 CFR 50.67, for calculating accident offsite doses and doses to control room personnel. The application and the supplements present the most limiting calculated dose consequences for the following events: main steam line break (MSLB) (inside containment), MSLB outside containment/feedwater line break, reactor coolant pump seized rotor/sheared shaft, control element assembly ejection accident, letdown line break, steam generator tube rupture, large break LOCA, small break LOCA, FHA, inadvertent atmospheric dump valve opening, and excess main steam flow with loss-of-offsite power.

The AST implementation amendment revised the design basis as described in the Updated Final Safety Analyses Report for FOL No. NPF-38. The effective date for this amendment is as of the date of issuance and to be implemented prior to restart from refueling outage 13 in spring of 2005 in order to update the design assumption regarding in-leakage, resolve concerns identified in GL 2003-01, and support the power uprate implementation.

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.

If you have any questions regarding this subject, please contact me at 301-415-1480.

Sincerely,

/RA/

N. Kalyanam, Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No.: 50-382

Enclosures: 1. Amendment No. 198 to NPF-38
2. Safety Evaluation

The AST implementation amendment revised the design basis as described in the Updated Final Safety Analyses Report for FOL No. NPF-38. The effective date for this amendment is as of the date of issuance and to be implemented prior to restart from refueling outage 13 in spring of 2005 in order to update the design assumption regarding in-leakage, resolve concerns identified in GL 2003-01, and support the power uprate implementation.

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ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 198
License No. NPF-38

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

   A. The application for amendment by Entergy Operations, Inc. (EOI) dated July 15, 2004, as supplemented by letters dated August 19, September 1, September 14, October 13, and October 19, 2004, 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 to approve changes to the Updated Final Safety Analysis Report (UFSAR), as set forth in the application for amendment by Entergy Operations, Inc., dated July 15, 2004, as supplemented by letters dated August 19, September 1, September 14, October 13, and October 19, 2004. Entergy Operations, Inc. shall update the UFSAR 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 prior to restart from the refueling outage 13 in the spring of 2005 in order to support the power uprate implementation. Implementation of the amendment is the incorporation into the UFSAR changes to the description of the facility as described in the licensee's application dated July 15, 2004, as supplemented by letters dated August 19, September 1, September 14, October 13, and October 19, 2004, and evaluated in the staff's Safety Evaluation dated

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by M.Webb for A.Howe/

Allen G. Howe, Chief, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: March 29, 2005

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 198 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 July 15, 2004 (Accession No. ML042020294), as supplemented by letters dated August 19 (ML042360712), September 1 (ML042470194), September 14 (ML042660243), October 13 (ML042890193), and October 19, 2004 (ML043010129), Entergy Operations, Inc. (Entergy or the licensee), requested an amendment for Waterford Steam Electric Station, Unit 3 (Waterford 3) that would allow Waterford 3 to use an alternative source term (AST), as permitted by 50.67 of Title 10 of the *Code of Federal Regulations,* for calculating offsite doses and doses to control room personnel. The supplements dated August 19, September 1, September 14, October 13, and October 19, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U. S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 19, 2004 (69 FR 51488).

Use of the AST would allow Entergy to revise the Waterford 3 Updated Final Safety Analyses Report (UFSAR) design basis. The revised design basis takes into account a proposed increase in core rated thermal power to 3716 thermal megawatts (MWt), also known as an extended power uprate (EPU). The licensee's EPU license amendment request was submitted by letter dated November 13, 2003, as supplemented, and is a separate licensing action.

To support the proposed implementation of an AST, the licensee performed radiological consequences analyses of the design basis accidents (DBAs) for Waterford 3, assuming operation at the uprated power. The licensee analyzed the dose consequences of the following DBAs:

- Main Steam Line Break (MSLB) Inside Containment
- MSLB Outside Containment/Feedwater Line Break
- Reactor Coolant Pump (RCP) Seized Rotor/Sheared Shaft
- Control Element Assembly Ejection Accident (REA)
- Letdown Line Break Outside Containment
- Steam Generator Tube Rupture (SGTR)
- Large Break Loss-of-Coolant Accident (LBLOCA)

- Small Break LOCA (SBLOCA)
- Fuel Handling Accident (FHA)

The licensee also analyzed the dose consequences of the following events:

- Inadvertent Atmospheric Dump Valve (ADV) Opening (IADVO event)
- Excess Main Steam Flow with Loss of Offsite Power (excess load event)

2.0 EVALUATION

In the analysis of DBAs, Entergy used the guidance in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," total effective dose equivalent (TEDE) radiological units and limits, and dose conversion factors from Federal Guidance Report (FGR) Nos. 11 and 12. The staff finds the use of FGR-11 and FGR-12 dose conversion factors acceptable as documented in RG 1.183. For additional detail on the inputs and assumptions Entergy used in its dose analyses, see the subject submittal and its supplements. The staff's review of the major inputs, assumptions, and calculational models used by the licensee in its dose analyses follows.

2.1 Source Terms and Radiological Consequences Analyses

2.1.1 Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

The NRC staff reviewed the radioactive source term associated with the AST and EPU to ensure the adequacy of the sources of radioactivity used by the licensee as input to calculations to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes. The review included the parameters used to determine: (1) the concentration of each radionuclide in the reactor coolant, (2) the fraction of fission product activity released to the reactor coolant, (3) concentrations of all radionuclides other than fission products in the reactor coolant, (4) leakage rates and associated fluid activity of all potentially radioactive water and steam systems, and (5) potential sources of radioactive materials in effluents that are not considered in the plant's UFSAR related to liquid waste management systems and gaseous waste management systems. The NRC's acceptance criteria for source terms are based on (1) 10 CFR Part 20, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas, (2) 10 CFR Part 50, Appendix I, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion, and (3) General Design Criterion (GDC) 60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in Standard Review Plan (SRP) Section 11.1.

Technical Evaluation

The core isotopic inventory is a function of the core power level. The reactor coolant activity concentrations are a function of the core power level, leakage from the fuel, radioactive decay, and removal by coolant purification systems. Entergy recalculated the maximum reactor coolant fission product activity concentration assuming 1 percent failed fuel, the expected reactor coolant concentration source terms for radioactive liquid and gaseous effluents for the higher proposed reactor power. The calculations use the methods and model outlines in Section 11.1.1.1 of the Waterford 3 UFSAR. Entergy also calculated the core isotopic inventory for the higher proposed reactor power for use in accident dose and equipment qualification dose evaluations.

The calculations assumed operation at a core power of 3735 MWt and for 590 effective full-power days. The assumed core power of 3735 MWt includes a power uncertainty of 0.5 percent. Other inputs and assumptions were unchanged from the original Waterford 3 design basis as specified in UFSAR Section 11.1. The NRC staff finds that the licensee has used the appropriate core power assumptions for the EPU. The NRC staff also finds that the EPU would not impact any of the other inputs and assumptions to the maximum coolant concentration calculations, and that continued use of the current UFSAR values is acceptable. The staff finds that Entergy has appropriately calculated the maximum reactor coolant fission product activity concentration for the EPU.

Entergy calculated the average reactor coolant fission product activity concentration using the American National Standards Institute (ANSI) 18.1 methodology (ANSI 18.1). The Waterford 3 original design basis was the ANSI N237 methodology. ANSI 18.1 is a revision to ANSI N237 and is a methodology that is acceptable to the staff.

The licensee compared the maximum and average reactor coolant concentrations using the EPU and Waterford 3 current design basis source terms in the UFSAR and found the current design basis source terms and resultant dose rates bound the uprate conditions.

The licensee also calculated that the EPU is expected to lead to a slight increase in circulating crud activities. The licensee used the current design basis methodology in UFSAR Section 11.1.2 to calculate these results for the EPU conditions. For circulating crud activities, the licensee continues to use the Waterford 3 current design basis. The staff finds the calculation of the circulating crud activities acceptable.

Entergy used the PWRGALE computer code to calculate the source terms for radioactive liquid and gaseous effluents for the uprated power. The Waterford 3 current design basis for these source terms includes ANSI N237 and the PWRGALE code methodology. The staff finds the PWRGALE code to be an acceptable methodology for this calculation.

Entergy used the Waterford 3 current design basis Technical Information Document (TID)-14844 core isotopic source term, adjusted for the increase in thermal power to 100.5 percent of the rated thermal power under EPU conditions (1.005x3716 MWt or 3735 MWt), to bound the current Waterford 3 power measurement uncertainty.

<u>Summary</u>

The NRC staff has reviewed the radioactive source term associated with the proposed AST and EPU and concludes that the proposed parameters and resultant composition and quantity of radionuclides are appropriate for the evaluation of the radioactive waste management systems. The NRC staff further concludes that the proposed radioactive source term meets the requirements of 10 CFR Part 20, 10 CFR Part 50, Appendix I, and GDC 60. Therefore, the NRC staff finds the proposed AST and EPU acceptable with respect to source terms.

2.1.2 Radiological Consequences of MSLB Inside Containment

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of an MSLB inside the containment. The NRC staff's review included (1) the sequence of events, models, and assumptions used by the licensee for the calculation of the radiological doses, (2) evaluation of the technical specifications (TSs) on the primary and secondary coolant iodine activities, and (3) determination of the amount of fuel damage and subsequent radiological source term. The NRC's acceptance criteria for the radiological consequences of an MSLB inside containment are based on (1) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident and (2) 10 CFR 50.67, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Section 15.0.1.

Technical Evaluation

Entergy's analysis of an MSLB inside the containment assumed that radioactive material in the primary coolant will be transported through primary-to-secondary leakage into both steam generators (SGs) and then into the containment via the break on the main steam line (MSL) on the affected SG. Release to the environment occurs through containment leakage. Steaming from the unaffected SG during plant cooldown is the pathway for additional release to the environment. The licensee stated that the fuel failure limits for this postulated accident are:

1. 2 percent core fuel cladding failure through departure from nucleate boiling (DNB) for the return to power MSLB inside containment with concurrent loss of offsite power (LOOP) (2 percent gap release),

2. 10 percent core fuel cladding failure through DNB for the MSLB inside containment with a concurrent LOOP, combined return to power and pre-trip power excursion results (10 percent gap release), or

3. 2 percent core fuel melt for the return to power MSLB inside containment without a concurrent LOOP (2 percent core activity release).

The MSLB outside containment does not result in fuel failure; therefore, the radiological source term is based on only the primary coolant activity. The licensee evaluated the MSLB outside containment separately, and staff's review of the MSLB outside containment is discussed below in Section 2.1.3.

The case with an MSLB inside containment without some concurrent LOOP results in 2 percent core fuel melt, gives the limiting radiological source term. This case also results in a slightly higher secondary steaming release due to the assumption that the RCPs remain running for 30 minutes after the MSLB.

The licensee's analysis of the MSLB inside containment accounts for two release pathways, containment leakage and secondary steaming. The dose consequences from the two pathways are added to give the total projected dose.

The licensee assumed that all the postulated release activity was released into the containment through the break within 1 second. The activity available for the containment release pathway is due to primary-to-secondary leakage from the reactor coolant system (RCS) into the affected SG equal to 540 gallons per day (gpd). The containment leakage release pathway model assumed that the containment was leaking at the design value of 0.5 percent volume per day for the first 24 hours, then leaked at 0.25 percent until the end of the accident, which was assumed to be 30 days. The licensee did not assume that iodine in the containment was removed through natural deposition or containment spray (CS) operation.

The secondary steaming pathway model assumed that releases are due to 150 gpd (which is twice the new SG leakage limit in TS 3.4.5.2 referenced in Entergy letter dated July 14, 2004) primary-to-secondary leakage from the RCS to the unaffected SG. Opening of the main steam safety valves (MSSVs) and use of the ADVs to cool the plant down to shutdown cooling conditions releases any available activity to the environment. The duration of the secondary steaming release was assumed to be 7.5 hours.

The licensee assumed the control room was isolated and the control room air conditioning system was placed into recirculation automatically by plant response to the accident. Filtered pressurization of the control room does not occur until 2 hours after the beginning of the accident by operator manual initiation. The licensee takes credit for the operators choosing the more favorable control room ventilation intake at the time of the initiation of pressurization (i.e., the intake with the lower radiation reading). The licensee assumed that the unfiltered inleakage into the control room was 100 cubic feet per minute (cfm) for the duration of the event, which bounds the measured tracer gas test results.

The staff reviewed the information provided in the licensee's AST submittal and supplements, and the Waterford 3 UFSAR and also performed an independent calculation that confirmed the licensee's dose results. Entergy's analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 2. The licensee's calculated dose results are given in Table 1. The licensee's calculated doses are within the SRP 15.0.1 radiological dose acceptance criteria for an MSLB with fuel damage. The TEDE criteria are 25 rem at the exclusion area boundary (EAB) for the worst 2 hours, 25 rem at the low population zone (LPZ) for the duration of the accident and 5 rem in the control room for the duration of the accident.

<u>Summary</u>

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of an MSLB inside containment and concludes that the licensee has adequately accounted for the effects of the proposed AST and EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating engineered safeguards features (ESFs) remain acceptable with respect to the radiological consequences of a postulated MSLB inside containment since the calculated TEDE at the EAB and the LPZ outer boundary and in the control room meet the exposure guideline values specified in 10 CFR 50.67 and GDC 19 and the dose acceptance criteria in SRP 15.0.1. Therefore, the NRC staff finds the licensee's proposed implementation of an AST and EPU acceptable with respect to the radiological consequences of MSLB accidents inside the containment.

2.1.3 Radiological Consequences of MSLB Outside Containment and Feedwater Line Break

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of an MSLB outside the containment. The NRC staff's review included (1) the sequence of events, models, and assumptions used by the licensee for the calculation of the radiological doses; (2) evaluation of the TSs on the primary and secondary coolant iodine activities; and (3) determination of reactor coolant iodine concentration corresponding to a preaccident iodine spike and a concurrent iodine spike. The NRC's acceptance criteria for the radiological consequences of an MSLB outside containment are based on (1) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident, and (2) 10 CFR 50.67, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Section 15.0.1.

Technical Evaluation

This dose analysis assumed that an MSL fails outside the containment and primary coolant is released to the environment. Entergy analyzed the MSLB outside containment and the feedwater line break through one dose analysis model, since the plant response, radiological source term, and accident progression characteristics are similar. Any differences between the two accident scenarios in steaming rates or release duration was modeled in the dose analysis with the value that will maximize the dose results.

As no fuel failure was predicted for either scenario, the radiological source term was based on the primary coolant activity. In accordance with RG 1.183 guidance, the two iodine spiking cases considered were: (1) pre-accident elevated iodine concentration equal to the TS maximum iodine concentration of 60 µCi/gm dose equivalent I-131 (DEI-131) and (2) accident-initiated iodine spiking of 500 times the iodine appearance rate that equates to the TS equilibrium iodine concentration limit of 1 µCi/gm DEI-131. The radioactive materials in reactor coolant are released to the environment through primary-to-secondary leakage into the SGs and then release through the break in the MSL until isolation and SG secondary coolant steaming through the unaffected SG ADV and MSSVs until the plant is cooled down enough for switchover to the residual heat removal (RHR) system at 36 hours.

The licensee assumed the control room was isolated and the control room air conditioning system was placed into recirculation automatically by plant response to the accident. Filtered pressurization of the control room is also assumed to initiate at the start of the accident. The licensee takes credit for the operators choosing the more favorable control room ventilation intake at 2 hours into the accident. The licensee assumed that the unfiltered inleakage into the control room was 100 cfm for the duration of the event, which bounds the measured tracer gas test results.

The staff reviewed the information provided in the licensee's AST submittal and supplements and the Waterford 3 UFSAR and also performed an independent calculation that confirmed the licensee's dose results. Entergy's analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 3. The licensee's calculated dose results are given in Table 1. The licensee's calculated doses from both analyzed iodine spiking cases are within the SRP 15.0.1 radiological dose acceptance criteria for an MSLB without fuel damage. For the pre-accident iodine spiking case, the TEDE criteria are 25 rem at the EAB for the worst 2 hours, 25 rem at the LPZ for the duration of the accident, and 5 rem in the control room for the duration of the accident. For the accident induced iodine spiking case, the TEDE criteria are 2.5 rem at the EAB for the worst 2 hours, 2.5 rem at the LPZ for the duration of the accident and 5 rem in the control room for the duration of the accident.

Summary

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of an MSLB outside containment and concludes that the licensee has adequately accounted for the effects of the proposed AST and EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated MSLB outside containment since the calculated TEDE at the EAB and the LPZ outer boundary and in the control room meet the exposure guideline values specified in 10 CFR 50.67 and GDC 19 and the dose acceptance criteria in SRP 15.0.1. Therefore, the NRC staff finds the licensee's proposed implementation of an AST and EPU acceptable with respect to the radiological consequences of MSLB accidents outside the containment.

2.1.4 Radiological Consequences of a Reactor Coolant Pump Locked Rotor Accident

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of a RCP locked rotor accident (LRA). The review included (1) determination of a need for a radiological consequences analysis and (2) the sequence of events, models, and assumptions used by the licensee for the calculation of radiological doses. The NRC's acceptance criteria for the radiological consequences of an RCP LRA are based on (1) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident and (2) 10 CFR 50.67, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP 15.0.1.

Technical Evaluation

Entergy's dose analysis of the RCP LRA assumed that the RCP was inoperable and loss of primary coolant circulation might result in as much as 15 percent of the core fuel rods experiencing departure from nucleate boiling (DNB) and subsequent cladding damage. The fission products in the gap of those damaged fuel rods were assumed released into the RCS coolant. Entergy used the non-LOCA gap fractions given as applicable to the control rod ejection accident in footnote 11 to Table 3 in RG 1.183. These values are larger than the gap fractions given in RG 1.183 Table 3 that would be applicable to the LRA. No fuel melting is assumed.

The radioactive materials in reactor coolant are transported through primary-to-secondary leakage of 150 gpd (which is twice the new SG leakage limit in TS 3.4.5.2 referenced in Entergy letter dated July 14, 2004) into each of the SGs. The activity is then assumed to be released to the environment through secondary coolant steaming through the ADVs and MSSVs until the plant is cooled down enough for switchover to the RHR system at 7.5 hours.

The licensee assumed the control room was isolated and the control room air conditioning system was placed into recirculation automatically by plant response to the accident. Filtered pressurization of the control room does not occur until 2 hours after the beginning of the accident by operator manual initiation. The licensee takes credit for the operators choosing the more favorable control room ventilation intake at the time of the initiation of pressurization (i.e., the intake with the lower radiation reading). The licensee assumed that the unfiltered inleakage into the control room was 150 cfm for the duration of the event, which bounds the measured tracer gas test results.

The staff reviewed the information provided in the licensee's AST submittal and supplements and the Waterford 3 UFSAR and also performed an independent calculation that confirmed the licensee's dose results. Entergy's analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 4. The licensee's calculated dose results are given in Table 1. The licensee's calculated doses are within the SRP 15.0.1 radiological dose acceptance criteria for an LRA. These TEDE criteria are 2.5 rem at the EAB for the worst 2 hours, 2.5 rem at the LPZ for the duration of the accident and 5 rem in the control room for the duration of the accident.

Summary

The NRC staff has evaluated the licensee's revised analyses for the radiological consequences of a RCP LRA and concludes that the licensee has adequately accounted for the effects of the proposed AST and EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated LRA since the calculated TEDE at the EAB and the LPZ outer boundary meet the dose acceptance criteria in SRP 15.0.1, are a small fraction of exposure guideline values specified in 10 CFR 50.67, and the control room TEDE is within the exposure guideline value specified in GDC 19. Therefore, the NRC staff finds the licensee's proposed implementation of an AST and EPU acceptable with respect to the radiological consequences of an LRA.

2.1.5 Radiological Consequences of a Control Element Ejection Accident

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of an REA. The NRC staff's review included the plant response to an REA and the calculation of radiological doses at the EAB and LPZ outer boundary and in the control room due to the releases resulting from an REA. The purpose of the NRC staff's review was to (1) ensure that plant's procedures for recovery from an REA and the plant's TSs are properly taken into account in computing the doses, and (2) compare the calculated doses against the appropriate guidelines. The NRC's acceptance criteria for the radiological consequences of an REA are based on (1) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident and (2) 10 CFR 50.67, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Section 15.0.1.

Technical Evaluation

Entergy's dose analysis of an REA considered two separate scenarios for fission product release to the environment: (1) primary-to-secondary leakage and secondary steaming until cold shutdown and (2) a break in the containment boundary and containment leakage for 30 days. The doses from the two release scenarios are not added together, but are considered separately.

For the source term, the licensee's dose analysis assumed 15 percent of the core fuel rods had experienced DNB and immediately released the fission products within the gap to the RCS coolant. Entergy used the non-loss-of-coolant accident (LOCA) gap fractions given as applicable to the REA in footnote 11 to Table 3 in RG 1.183. No fuel melting was assumed.

For the secondary steaming scenario, the radioactive materials in the reactor coolant are transported through primary-to-secondary leakage of 150 gpd (which is twice the new SG leakage limit in TS 3.4.5.2 referenced in Entergy letter dated July 14, 2004) into each of the SGs. The activity is then assumed to be released to the environment through secondary coolant steaming through the ADVs and MSSVs until the plant is cooled down enough for switchover to the RHR system at 7.5 hours.

For the containment leakage release scenario, Entergy assumed that all the activity available for release was released into the containment through the break immediately. The containment leakage release pathway model assumed that the containment was leaking at the design value of 0.5 percent volume per day for the first 24 hours, then leaked at 0.25 percent until the end of the accident, which was assumed to be 30 days. The licensee did not assume that iodine in the containment was removed through CS operation. Natural deposition of elemental and particulate iodine was assumed, and the licensee assumptions followed guidance in RG 1.183.

The licensee assumed the control room was isolated and the control room air conditioning system was placed into recirculation automatically by plant response to the accident. Filtered pressurization of the control room does not occur until 2 hours after the beginning of the accident by operator manual initiation. The licensee takes credit for the operators choosing the more favorable control room ventilation intake at the time of the initiation of pressurization (i.e., the intake with the lower radiation reading). The licensee assumed that the unfiltered inleakage into the control room was 150 cfm for the duration of the event, which bounds the measured tracer gas test results.

The staff reviewed the information provided in the licensee's AST submittal and supplements and the Waterford 3 UFSAR and also performed an independent calculation that confirmed the licensee's dose results. Entergy's analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 5. The licensee's calculated dose results are given in Table 1. The licensee's calculated doses from either release scenario are within the SRP 15.0.1 radiological dose acceptance criteria for a element ejection accident. These TEDE criteria are 6.3 rem at the

EAB for the worst 2 hours, 6.3 rem at the LPZ for the duration of the accident and 5 rem in the control room for the duration of the accident.

Summary

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a REA and concludes that the licensee has adequately accounted for the effects of the proposed AST and EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated REA since the calculated TEDE at the EAB and the LPZ outer boundary meet the dose acceptance criteria in SRP 15.0.1, are a small fraction of exposure guideline values specified in 10 CFR 50.67, and the control room TEDE is within the exposure guideline value specified in GDC 19. Therefore, the NRC staff finds the licensee's proposed implementation of an AST and EPU acceptable with respect to the radiological consequences of an REA.

2.1.6 Radiological Consequences of a Letdown Line Break Outside Containment

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of failures outside the containment of letdown system piping. The NRC staff's review included (1) the identification of the letdown lines postulated to fail and the isolation provisions for these lines, (2) the failure scenario, (3) the models and assumptions for the calculation of the radiological doses for the postulated failure, and (4) an evaluation of the primary coolant iodine activity, including the effects of a concurrent iodine spike, and the TSs for the reactor coolant iodine activity. The NRC's acceptance criteria for the radiological consequences of the failure of small lines carrying primary coolant outside containment are based on (1) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident, and (2) GDC 55, insofar as it establishes isolation requirements for small-diameter lines connected to the primary system that form the basis of complying with 10 CFR 50.67. Specific review criteria are contained in SRP Sections 15.0.1 and 15.6.2 and further guidance in Safety Guide 11 (RG 1.11).

Technical Evaluation

The letdown line is the largest piping that carries RCS fluid outside containment. A rupture of the letdown line outside containment provides a release path for the primary coolant to the outside environment. Entergy performed a revised analysis for the letdown line break outside containment for EPU conditions based on current licensing basis DBA dose analyses as documented in Waterford 3 UFSAR Chapter 15.6.3.1.

Entergy's analysis assumed that no fuel failure results from the letdown line break and the radioactivity in the RCS was initially at the long-term TS limit of 1 μCi/gm DEI-131, as defined in the Waterford 3 TS. The accident was assumed to cause the iodine concentration to spike by a factor of 500 times the equilibrium iodine appearance rate. A total of 78,000 lbm of RCS fluid was assumed released through the break. Additional RCS radioactivity was assumed released to the environment through SG tube leakage and secondary system steaming to cool down the plant.

The licensee assumed filtered pressurization of the control room initiated at the beginning of the event. The licensee takes credit for the operators choosing the more favorable control room ventilation intake at 2 hours. The licensee assumed that the unfiltered inleakage into the control room was 150 cfm for the duration of the event, which bounds the measured tracer gas test results.

The staff reviewed the information provided in the licensee's EPU submittal and supplements and the Waterford 3 UFSAR and also performed an independent calculation that confirmed the licensee's dose results. Entergy's analysis used assumptions and inputs that follow the guidance in SRP 15.6.2 and similar accidents in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 6. The licensee's calculated dose results are given in Table 1. Although there are no specific dose acceptance criteria given in SRP 15.0.1 for the letdown line break, the acceptance limits previously established for Waterford 3 are 10 CFR Part 100 limits for the pre-accident iodine spike and a small fraction (10 percent) of 10 CFR Part 100 limits for the accident induced iodine spike. The staff uses the same logic for application to the TEDE criterion in 10 CFR 50.67. Therefore, for the pre-accident iodine spiking case, the TEDE criteria are 25 rem at the EAB for the worst 2 hours, 25 rem at the LPZ for the duration of the accident and 5 rem in the control room for the duration of the accident. For the accident induced iodine spiking case, the TEDE criteria are 2.5 rem at the EAB for the worst 2 hours, 2.5 rem at the LPZ for the duration of the accident and 5 rem in the control room for the duration of the accident. Entergy's analysis results meet these dose acceptance criteria.

Summary

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a letdown line break outside the containment and concludes that the licensee has adequately accounted for the effects of the proposed AST and EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated letdown line break since the calculated TEDE at the EAB and the LPZ outer boundary are substantially below the exposure guideline values of 10 CFR 50.67 and the control room TEDE is within the exposure guideline value specified in GDC 19. Therefore, the NRC staff finds the licensee's proposed implementation of an AST and EPU acceptable with respect to the radiological consequences of a letdown line break.

2.1.7 Radiological Consequences of Steam Generator Tube Rupture

Regulatory Evaluation

The NRC staff reviewed the analysis of the radiological consequences of a postulated SGTR. The NRC staff's review included (1) a review of the sequence of events and plant procedures for recovery from the accident to ensure that the most severe case of radioactive releases has been considered, (2) a review of the models and assumptions for the calculation of the radiological doses for the postulated accident, (3) an evaluation of the TSs on the primary and secondary coolant iodine activity concentration, and (4) an evaluation of the radiological consequences of an SGTR concurrent with a LOOP and the most limiting single failure. The

NRC staff's review included two cases for the reactor coolant iodine concentration corresponding to a preaccident iodine spike and a concurrent iodine spike. The NRC's acceptance criteria for the radiological consequences of an SGTR are based on (1) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 TEDE for the duration of the accident and (2) 10 CFR 50.67, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Section 15.0.1.

Technical Evaluation

Entergy submitted a revised analysis of the SGTR radiological consequences by letter dated October 18, 2004 (ML042940577). This SGTR analysis used the CENTS code to calculate the projected steam mass release and timing. Entergy's revised radiological consequences analysis also accounted for a second release late in the accident caused by the reactor operators opening the ADVs to continue to cool down the plant. Entergy's revised SGTR analysis was based on the current licensing basis DBA dose analyses as documented in Waterford 3 UFSAR Chapter 15.6.3.2.

Entergy's radiological consequences analysis assumed two radiological source term cases; an accident induced iodine spike (which the licensee denoted as GIS) and a pre-existing iodine spike (which the licensee denoted as PIS). The accident induced iodine spike case assumed the primary coolant activity was initially at the long-term TS limit of 1.0 μCi/gm DEI-131. The accident was assumed to cause the iodine concentration to spike by a factor of 500 times the equilibrium iodine appearance rate. The pre-existing iodine spike case assumed the primary coolant iodine concentration was at an increased value of 60 μCi/gm DEI-131. This is the TS limit for full power operation following an iodine spike for up to 48 hours. A total of 325,702 lbm of primary coolant was assumed to pass through the ruptured SG tubes and available for release to the outside environment by steaming through the ruptured SG ADV.

The licensee assumed the control room was isolated and the control room air conditioning system was placed into recirculation automatically by plant response to the accident. Filtered pressurization of the control room does not occur until 2 hours after the beginning of the accident by operator manual initiation. The licensee takes credit for the operators choosing the more favorable control room ventilation intake at the time of the initiation of pressurization (i.e., the intake with the lower radiation reading). The licensee assumed that the unfiltered inleakage into the control room was 100 cfm for the duration of the event, which bounds the measured tracer gas test results.

Entergy calculated effective control room atmospheric dispersion factors (χ/Qs) to account for the different amounts of fission products in steam released to the environment from each of the two SG ADVs, and their relative dispersion to each control room air intake. The relative fission product release contribution from each ADV was based directly on the curie releases calculated in the CENTS analysis documented in Section 2.13.6.3.2 of the EPU license amendment request dated November 13, 2003. The staff has evaluated the scaling of the control room χ/Qs to account for the relative release from each SG ADV and finds the calculation of effective factors acceptable. The staff has evaluated and found acceptable the base control room χ/Qs for releases from the ADVs as discussed below in Section 2.3. The effective control room χ/Q values used in this AST SGTR dose analysis are only applicable to the particular release characteristics described in the submittal. The control room χ/Q values to be used in any revision to the SGTR dose analysis in the future should consider the impact of the change in steaming rates and/or activity release on the control room effective χ/Q for releases from the ADVs.

The staff reviewed the information provided in the licensee's AST submittal and supplements and the Waterford 3 UFSAR and also performed an independent calculation that confirmed the licensee's dose results. Entergy's analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 7. The licensee's calculated dose results are given in Table 1. The licensee's calculated doses are within the SRP 15.0.1 radiological dose acceptance criteria for an SGTR. For the pre-accident iodine spiking case, the TEDE criteria are 25 rem at the EAB for the worst 2 hours, 25 rem at the LPZ for the duration of the accident and 5 rem in the control room for the duration of the accident. For the accident induced iodine spiking case, the TEDE criteria are 2.5 rem at the EAB for the worst 2 hours, 2.5 rem at the LPZ for the duration of the accident and 5 rem in the control room for the duration of the accident.

<u>Summary</u>

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of an SGTR and concludes that the licensee has adequately accounted for the effects of the proposed AST and EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of an SGTR accident since the calculated TEDE at the EAB and the LPZ outer boundary do not exceed the exposure guideline values of 10 CFR 50.67 (assuming a preaccident iodine spike) and are a small fraction of the 10 CFR 50.67 values for the concurrent iodine spike. Additionally, the control room TEDE is within the exposure guideline value specified in GDC 19. Therefore, the NRC staff finds the licensee's proposed AST and EPU acceptable with respect to the radiological consequences of an SGTR.

2.1.8 <u>Radiological Consequences of a Design-Basis Large Break Loss-of-Coolant Accident</u>

<u>Regulatory Evaluation</u>

The NRC staff reviewed the analyses of the radiological consequences of a design-basis LBLOCA. The review included a summary review of the doses from the hypothetical design-basis LBLOCA and a specific review of the doses from containment leakage and leakage from ESF components outside containment that contribute to the total LBLOCA doses. The NRC staff's review also included (1) the methodology and results of calculations of the radiological consequences resulting from containment and ESF component leakage following a hypothetical LOCA and (2) an assessment of the containment with respect to the assumptions and the values of input parameters for the dose calculations. The NRC staff's calculations are based on pertinent information in the UFSAR and considered the NRC staff's evaluation of dose-mitigating ESFs. The NRC's acceptance criteria for the radiological consequences of a design-basis LBLOCA are based on (1) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 TEDE for the duration of the accident and (2) 10 CFR 50.67, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Section 15.0.1.

Technical Evaluation

Entergy performed a revised analysis of the radiological consequences of an LBLOCA for the EPU conditions using the guidance on source term and activity release in RG 1.183. The revised analysis assumed a core thermal power of 3735 MWt, which is 100.5 percent of the proposed uprated power of 3716 MWt. Entergy evaluated three release pathways in the LBLOCA dose analysis: (1) leakage to the reactor auxiliary building serviced by the controlled ventilation area system (CVAS); (2) leakage to the secondary containment that is serviced by the shield building ventilation system (SBVS); and (3) leakage from the containment directly to the environment. The first two release pathways are filtered, but the third is not. Additionally, emergency core cooling system (ECCS) components outside containment were assumed to leak 0.5 gallons per minute of containment sump water into areas serviced by the CVAS, starting when the ECCS goes into recirculation mode. A portion (10 percent) of this leaked sump water was assumed to flash to steam, causing the radioactive iodine in that portion to become airborne and available for release to the outside environment.

Entergy assumed 80 percent of the containment volume is subject to fission product removal by the CSs. Consistent with RG 1.183, the licensee's analysis assumed a mixing rate due to natural convection between the sprayed and unsprayed regions of the containment equal to two turnovers of the unsprayed volume per hour. The licensee did not credit forced mixing from the containment fan coolers, of which at least one is expected be available. In the sprayed region of containment, the licensee assumed only particulate iodine was removed by the sprays, with removal coefficients calculated based on the guidance in SRP Section 6.5.2. In the unsprayed region of containment, the licensee assumed natural deposition of elemental and particulate iodine. The particulate iodine natural deposition coefficients are based on the model in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," which is incorporated as the Powers 10 percent aerosol decontamination model in the dose analysis code RADTRAD, used by the licensee. The elemental iodine natural deposition coefficients are calculated based on the guidance in SRP 6.5.2. The staff has identified these as appropriate iodine and aerosol spray and natural deposition removal models in RG 1.183.

The containment leakage release pathway model assumed that the containment was leaking at the design value of 0.5 percent volume per day for the first 24 hours, then leaked at 0.25 percent until the end of the accident, which was assumed to be 30 days.

RG 1.183 gives assumptions for the amount and chemical form of radionuclides released to the containment for a postulated LOCA. These assumptions are applicable if the licensee can show that the sump pH is controlled at values of seven or greater. If the sump pH falls below seven, iodine re-evolution should be considered. Entergy performed an evaluation of the post LOCA sump pH to show that the RG 1.183 assumptions are applicable. The staff review of the licensee's evaluation for containment sump pH and iodine spray coefficients is in Section 2.2 of this SE.

The licensee assumed the control room was isolated and the control room air conditioning system was placed into recirculation automatically by plant response to the accident. Filtered pressurization of the control room does not occur until 2 hours after the beginning of the accident by operator manual initiation. The licensee takes credit for the operators choosing the more favorable control room ventilation intake at the time of the initiation of pressurization (i.e., the intake with the lower radiation reading). The licensee assumed that the unfiltered inleakage into the control room was 100 cfm for the duration of the event, which bounds the measured tracer gas test results.

Entergy also calculated the dose to the control room operators due to gamma shine from radioactive material trapped by the control room emergency air recirculation system, SBVS and CVAS filters. The analysis assumptions were the same as described above for the LBLOCA inhalation and submersion dose analyses, with the following exceptions:

1. The containment leakage assumed to be directly released to the environment (6 percent) is assumed instead to be directed to the CVAS, increasing the trapped radioactivity on the CVAS filters.

2. The deposition on the control room recirculation filters assumed a higher 200 cfm unfiltered inleakage value.

3. The assumed building volume for the CVAS filter deposition was conservatively modeled and no deposition was assumed in the CVAS or shield building.

4. Filter efficiencies of 100 percent were assumed to maximize the amount of radioactive material deposited on the filters.

5. After 24 hours, the flashing fraction for the ECCS leakage is less than the 10 percent used in the above described analysis.

The first four differences are conservative, while the ECCS leakage flashing fraction is less conservative than the value used in the offsite and control room airborne dose analyses. The licensee provided additional information to try to justify the 2 percent flashing fraction it had based on calculation of the enthalpy in the leaked fluid. In RG 1.183, the staff states that if the temperature of the leakage is less than 212 EF or the calculated flashing fraction is less than 10 percent, the amount of iodine that becomes airborne should be assumed to be 10 percent of the total iodine in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates. The 10 percent flashing fraction assumption accounts for the uncertainty in the release of iodine from this leaked ECCS fluid. Several aspects of what happens to the leaked fluid once it leaves the ECCS components are unknown. For example, the leaked fluid may be sprayed in a fine mist or may drip out of the ECCS components, the fluid on the floor may evaporate to dryness, or the fluid may collect in floor drains with an unknown pH. Therefore, the amount of iodine becoming airborne through these mechanisms is uncertain.

The staff does not find the information on containment sump pH and area ventilation rates supplied by the licensee in its September 1, 2004, letter sufficient to find a flashing fraction of less than 10 percent for the ECCS leakage acceptable. The licensee's information did not fully address the staff's concerns about the uncertainty in the iodine release from the ECCS leaked fluid from the conditions that the leaked fluid may experience. However, the licensee does assume a 10 percent flashing fraction for the first 24 hours, and there is additional conservatism in the filter shine dose analysis as noted above in the filter shine analysis, which help compensate for the lower flashing fraction after 24 hours. The licensee does use the appropriate 10 percent flashing fraction assumption for the duration of the release in its offsite and control room inhalation and submersion dose analyses. The staff does not find the assumption of an ECCS leakage flashing fraction and airborne iodine of 2 percent acceptable. The staff does find the control room filter shine dose analysis results, including the shine dose due to ECCS leakage, acceptable because of the stated compensating conservatism in the overall filter shine dose analyses. The staff recommends that the ECCS flashing fraction be either increased to 10 percent or the lower value justified when the licensee revises the LBLOCA control room filter shine dose analyses in the future.

The staff reviewed the information provided in the licensee's AST submittal and supplements and the Waterford 3 UFSAR and also performed an independent calculation that confirmed the licensee's dose results. Entergy's analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 8. The licensee's calculated dose results are given in Table 1. The licensee's calculated doses are within the SRP 15.0.1 radiological dose acceptance criteria for a LOCA. These TEDE criteria are 25 rem at the EAB for the worst 2 hours, 25 rem at the LPZ for the duration of the accident and 5 rem in the control room for the duration of the accident.

Summary

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a design-basis LBLOCA and concludes that the licensee has adequately accounted for the effects of the proposed AST and EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a design-basis LBLOCA since the calculated TEDE at the EAB and the LPZ outer boundary do not exceed the exposure guideline values of 10 CFR 50.67 and the control room TEDE is within the exposure guideline value specified in GDC 19. Therefore, the NRC staff finds the licensee's proposed AST and EPU acceptable with respect to the radiological consequences of a design-basis LBLOCA.

2.1.9 Radiological Consequences of a Small Break Loss-of-Coolant Accident

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of an SBLOCA. The NRC staff's review included (1) review of the methodology and results of calculations of the radiological consequences resulting from containment leakage and primary-to-secondary leakage with secondary side steaming and (2) an assessment of the containment with respect to the assumptions and the values of input parameters for the dose calculations. The NRC staff's calculations are based on pertinent information in the UFSAR and considered the NRC staff's evaluation of dose-mitigating ESFs. The NRC's acceptance criteria for the radiological consequences of an SBLOCA are based on (1) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 TEDE for the duration of the accident and (2) 10 CFR 50.67, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Section 15.0.1.

Technical Evaluation

Entergy performed a revised analysis of the radiological consequences of an SBLOCA for the EPU conditions based on implementation of an AST. The revised analysis assumed a core thermal power of 3735 MWt, which is 100.5 percent of the proposed uprated power of 3716 MWt. Entergy's dose analysis of an SBLOCA considered two separate scenarios for fission product release to the environment: (1) primary-to-secondary leakage and secondary steaming until cold shutdown and (2) a break in the containment boundary and containment leakage for 30 days. The doses from the two release scenarios are not added together, but are considered separately

For the source term, the licensee's dose analysis assumed 100 percent of the core fuel rods experience cladding failure and immediately release the fission products within the gap to the RCS coolant. Entergy used the LOCA gap fractions given in RG 1.183 Table 2. No fuel melting was assumed.

For the secondary steaming scenario, the radioactive materials in the reactor coolant are transported through primary-to-secondary leakage of 75 gpd (which is the new SG leakage limit in TS 3.4.5.2 referenced in Entergy letter dated July 14, 2004) into each of the SGs, which is the proposed TS limit for the EPU. The activity is then assumed to be released to the environment through secondary coolant steaming through the ADVs and MSSVs until the plant is cooled down enough for switchover to the RHR system at 7.5 hours.

For the containment leakage release scenario, Entergy assumed that all the activity available for release was released into the containment through the break immediately. The containment leakage release pathway model assumed that the containment was leaking at the design value of 0.5 percent volume per day for the first 24 hours, then leaked at 0.25 percent until the end of the accident, which was assumed to be 30 days. As in the LBLOCA, the licensee assumed that iodine in the containment was removed through CS operation and natural deposition of elemental and particulate iodine occurs in the unsprayed region of the containment. The licensee's assumptions on iodine and particulate removal followed guidance in RG 1.183 and other applicable guidance as discussed above in Section 2.1.8 on the LBLOCA.

The licensee assumed the control room was isolated and the control room air conditioning system was placed into recirculation automatically by plant response to the accident. Filtered pressurization of the control room does not occur until 2 hours. The licensee takes credit for the operators choosing the more favorable control room ventilation intake at the time of the initiation of pressurization. The licensee assumed that the unfiltered inleakage into the control room was 100 cfm for the duration of the event for the containment release scenario, but assumed a reduction to 65 cfm upon pressurization for the secondary steaming release scenario.

The staff reviewed the information provided in the licensee's AST submittal and supplements and the Waterford 3 UFSAR and also performed an independent calculation that confirmed the licensee's dose results. Entergy's analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 9. The licensee's calculated dose results are given in Table 1. There are no specific dose acceptance criteria given in SRP 15.0.1 for the SBLOCA, but the LOCA dose acceptance criteria apply as for the LBLOCA. The licensee's calculated doses are within the

SRP 15.0.1 radiological dose acceptance criteria for a LOCA. These TEDE criteria are 25 rem at the EAB for the worst 2 hours, 25 rem at the LPZ for the duration of the accident and 5 rem in the control room for the duration of the accident.

Summary

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a design-basis SBLOCA and concludes that the licensee has adequately accounted for the effects of the proposed AST and EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a design-basis SBLOCA since the calculated TEDE at the EAB and the LPZ outer boundary do not exceed the exposure guideline values of 10 CFR 50.67 and the control room TEDE is within the exposure guideline value specified in GDC 19. Therefore, the NRC staff finds the licensee's proposed AST and EPU acceptable with respect to the radiological consequences of a design-basis SBLOCA.

2.1.10 Radiological Consequences of Fuel Handling Accidents

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of a postulated FHA. The purpose of this review was to evaluate the adequacy of system design features and plant procedures provided for the mitigation of the radiological consequences of accidents that involve damage to spent fuel. Such accidents include the dropping of a single fuel assembly and handling tool or a heavy object onto other spent fuel assemblies. Such accidents may occur inside the containment, along the fuel transfer canal, and in the fuel building. The NRC staff's review included (1) the sequence of events, models, and assumptions used by the licensee for the calculation of radiological doses, (2) the adequacy of the ESFs provided for the purpose of mitigating potential accident doses, and (3) the containment ventilation system with respect to its function as a dose-mitigating ESF system, including the radiation detection system on the containment purge/vent lines for those plants that will vent or purge the containment during fuel handling operations. The NRC's acceptance criteria for the radiological consequences of FHAs are based on (1) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident, (2) GDC 61, insofar as it requires that systems that contain radioactivity be designed with appropriate containment, confinement, and filtering systems, and (3) 10 CFR 50.67, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Section 15.0.1.

Technical Evaluation

Entergy performed a revised analysis of the radiological consequences of an FHA for the AST and EPU using the guidance in RG 1.183. The licensee's analysis assumed that the dropping of a fuel assembly causes 60 fuel rods to be damaged with instantaneous release of the fission products in the gap of those rods. The damaged fuel was assumed to have undergone 72 hours of radioactive decay in accordance with Waterford 3 TS 3/4.9.3. The licensee assumed the fraction of fission products in the fuel rod gap was more conservative than the standard assumption in RG 1.183 for both I-131 and Kr-85. Entergy identified four possible release locations for the FHA: (1) the plant stack, (2) the containment equipment hatch, (3) the personnel airlock doors, and (4) the fuel handling building (FHB) truck bay. Entergy determined that the plant stack release location would give the highest calculated doses because of the atmospheric dispersion assumptions associated with the location. No FHB or containment holdup, filtration, or recirculation is credited in the analysis, and the licensee assumes an effective iodine decontamination factor of 200 for the overlying pool.

The licensee assumed the control room was isolated and the control room air conditioning system was placed into recirculation automatically by plant response to the accident. Filtered pressurization of the control room is assumed initiated at the start of the event. The licensee does not take credit for the operators choosing the more favorable control room ventilation intake at the time of the initiation of pressurization. The licensee assumed that the unfiltered inleakage into the control room was 200 cfm for the duration of the event, which bounds the measured tracer gas test results. The staff finds that these assumptions maximize the control room dose.

The staff reviewed the information provided in the licensee's AST submittal and supplements and the Waterford 3 UFSAR and also performed an independent calculation that confirmed the licensee's dose results. The licensee's analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 10. The licensee's calculated dose results are given in Table 1. The licensee's calculated doses are within the SRP 15.0.1 radiological dose acceptance criteria for an FHA. These TEDE criteria are 6.3 rem at the EAB for the worst 2 hours, 6.3 rem at the LPZ for the duration of the accident, and 5 rem in the control room for the duration of the accident.

Summary

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of FHAs and concludes that the licensee has adequately accounted for the effects of the proposed AST and EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated FHA since the calculated TEDE at the EAB and the LPZ boundary are well within the exposure guideline values of 10 CFR 50.67 and the control room TEDE is within the exposure guideline value specified in GDC 19. Therefore, the NRC staff finds the licensee's proposed AST and EPU acceptable with respect to the offsite radiological consequences of FHAs.

2.1.11 Radiological Consequences of an Inadvertent Atmospheric Dump Valve Opening

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of an IADVO event. The NRC staff's review included (1) determining a need for a design-basis radiological analysis, (2) sequence of events, models, and assumptions used by the licensee for the calculation of the radiological doses, and (3) comparing the calculated doses to exposure guidelines to determine the acceptability of the EAB and LPZ outer boundary distances and to confirm the adequacy of ESFs provided for the purpose of mitigating potential doses from DBAs, including the effects on control room habitability. The NRC's acceptance criteria for the radiological consequences of the IADVO event are based on (1) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of five TEDE for the duration of the accident and (2) 10 CFR 50.67, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 15.0.1 and 15.1.1-15.1.4.

Technical Evaluation

The plant response and release scenario for an IADVO event is similar to that for the MSLB outside containment accident. As no fuel failure was predicted for the IADVO event, the radiological source term was based on the primary coolant activity. Two iodine spiking cases were considered: (1) pre-accident elevated iodine concentration equal to the TS maximum iodine concentration of 60 µCi/gm DEI-131; and (2) accident-initiated iodine spiking of 500 times the iodine appearance rate that equates to the TS equilibrium iodine concentration limit of 1 µCi/gm DEI-131. The radioactive materials in reactor coolant are released to the environment through (1) primary-to-secondary leakage into the SGs and then released through the break in the MSL until isolation and (2) SG secondary coolant steaming through the unaffected SG ADV and MSSVs until the plant is cooled down enough for switchover to the RHR system at 36 hours.

The licensee assumed the control room was isolated and the control room air conditioning system was placed into recirculation automatically by plant response to the accident. Filtered pressurization of the control room is assumed initiated at the start of the event. The licensee takes credit for the operators choosing the more favorable control room ventilation intake at 2 hours into the event. The licensee assumed that the unfiltered inleakage into the control room was 100 cfm for the duration of the event, which bounds the measured tracer gas test results.

The staff reviewed the information provided in the licensee’s AST submittal and supplements and the Waterford 3 UFSAR and also performed an independent calculation that confirmed the licensee’s dose results. The licensee’s analysis used assumptions and inputs that follow the guidance for similar accidents in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 11. The licensee’s calculated dose results are given in Table 1. There are no specific dose acceptance criteria in SRP 15.0.1 or SRP 15.1.1-15.1.4 for an IADVO event. The staff considers this event to be similar to an MSLB outside containment accident without fuel failure and will use the SRP 15.0.1 dose acceptance criteria for the MSLB. For the pre-accident iodine spiking case, the TEDE criteria are 25 rem at the EAB for the worst 2 hours, 25 rem at the LPZ for the duration of the accident and 5 rem in the control room for the duration of the accident. For the accident induced iodine spiking case, the TEDE criteria are 2.5 rem at the EAB for the worst 2 hours, 2.5 rem at the LPZ for the duration of the accident and 5 rem in the control room for the duration of the accident. The licensee’s calculated doses are within these acceptance criteria.

Summary

The NRC staff has evaluated the licensee’s discussion of the radiological consequences of an IADVO event and concludes that the licensee has adequately accounted for the effects of the proposed AST and EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of an IADVO event. Therefore, the NRC staff finds the licensee's proposed AST and EPU acceptable with respect to the radiological consequences of an IADVO event.

2.1.12 Radiological Consequences of Excess Main Steam Flow with LOOP

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of excess main steam flow with concurrent LOOP, which will be referred to below as the excess load event. The NRC staff's review included (1) determining a need for a design-basis radiological analysis, (2) sequence of events, models, and assumptions used by the licensee for the calculation of the radiological doses, and (3) comparing the calculated doses to exposure guidelines to determine the acceptability of the EAB and LPZ outer boundary distances and to confirm the adequacy of ESFs provided for the purpose of mitigating potential doses from DBAs, including the effects on control room habitability. The NRC's acceptance criteria for the radiological consequences of the excess load event are based on (1) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 TEDE for the duration of the accident and (2) 10 CFR 50.67, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 15.0.1 and 15.1.1-15.1.4.

Technical Evaluation

The licensee's modeling for the excess load event is similar to its analysis of an LRA. Fuel failure of 8 percent of the core was predicted for the excess load event. The fission products in the gap of those damaged fuel rods were assumed released into the RCS coolant. Entergy used the non-LOCA gap fractions given as applicable to the REA in footnote 11 to Table 3 in RG 1.183. These values are larger than the gap fractions given in RG 1.183 Table 3 that would be applicable to the excess load event. No fuel melting is assumed.

The radioactive materials in reactor coolant are transported through primary-to-secondary leakage of 150 gpd (which is twice the new SG leakage limit in TS 3.4.5.2 referenced in Entergy letter dated July 14, 2004) into each of the SGs. The activity is then assumed to be released to the environment through secondary coolant steaming through the ADVs and MSSVs until the plant is cooled down enough for switchover to the RHR system at 7.5 hours.

The licensee assumed the control room was isolated and the control room air conditioning system was placed into recirculation automatically by plant response to the accident. Filtered pressurization of the control room does not occur until 2 hours after the beginning of the accident by operator manual initiation. The licensee takes credit for the operators choosing the more favorable control room ventilation intake at the time of the initiation of pressurization (i.e., the intake with the lower radiation reading). The licensee assumed that the unfiltered inleakage into the control room was 150 cfm for the duration of the event, which bounds the measured tracer gas test results.

The staff reviewed the information provided in the licensee's AST submittal and supplements and the Waterford 3 UFSAR and also performed an independent calculation that confirmed the licensee's dose results. The licensee's analysis used assumptions and inputs that follow the guidance for similar accidents in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 12. The licensee's calculated dose results are given in Table 1. There are no specific dose acceptance criteria in SRP 15.0.1 or SRP 15.1.1-15.1.4 for an excess load event. The staff considers this event to be similar to an LRA with fuel failure and will use the dose acceptance for the LRA in SRP 15.0.1. The TEDE criteria are 2.5 rem at the EAB for the worst 2 hours, 2.5 rem at the LPZ for the duration of the accident, and 5 rem in the control room for the duration of the accident. The licensee's calculated doses are within these acceptance criteria.

Summary

The NRC staff has evaluated the licensee's discussion of the radiological consequences of the excess load event and concludes that the licensee has adequately accounted for the effects of the proposed AST and EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of the excess load event. Therefore, the NRC staff finds the licensee's proposed AST and EPU acceptable with respect to the radiological consequences of an excess load event.

2.2 Containment Sump pH and Iodine Spray Removal Coefficients

Regulatory Evaluation

Implementation of an AST involves re-analyzing DBAs according to 10 CFR 50.67 and applying for a license amendment under 10 CFR 50.90. The license amendment application for Waterford Unit 3 credits the use of trisodium phosphate (TSP) to minimize iodine re-evolution following a postulated LOCA. Regulatory guidance for the implementation of the AST is provided in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

Technical Evaluation

2.2.1 Containment Sump pH

A variety of acids and bases are produced in containment after a LOCA. The pH value of the containment sump will depend on the chemical species dissolved in the containment sump water. The licensee identified the following chemical species that are introduced into the containment sump in a post-LOCA environment: hydriodic acid (HI), nitric acid (HNO_3), hydrochloric acid (HCl), and cesium hydroxide (CsOH). CsOH and HI enter the containment directly from the RCS. HCl is produced by radiolytic decomposition of cable jacketing and HNO_3 is synthesized in the radiation field existing in the containment. The resultant containment sump pH will depend on their relative concentrations and on the buffering action of TSP. Maintaining sump water in an alkaline condition is needed for preventing dissolved radioactive iodine from being released to the containment atmosphere during the recirculation CS injection. Most of the iodine leaves the damaged core in an ionic form, which is readily dissolved in the sump water. However, in an acidic environment, some of it becomes converted into elemental form, which is much less soluble, causing re-evolution of iodine to the containment atmosphere. Per NUREG-1465, "Accident Source Terms for Light-Water Nuclear

Power Plants," the iodine entering the containment is at least 95 percent cesium iodide (CsI) with the remaining 5 percent elemental and organic iodide plus HI, with not less than 1 percent of each iodine and HI. In order to prevent release of elemental iodine to the containment atmosphere after a LOCA, the sump pH has to be maintained equal to or higher than 7.

After a LOCA, the containment sump is mostly filled with water coming from the systems containing boric acid: refueling water storage pool (RWSP), safety injection tanks (SITs), boric acid mixing tank (BAMT) and RCS. This in effect will cause the sump water pH to become acidic. In order to keep the pH above 7, Waterford 3 uses TSP as a buffer to maintain the pH above 7 for the 30-day period after a LOCA.

The licensee utilized the STARpH 1.04 code to determine the containment sump pH 30 days after a LOCA. The code performs the calculation in three consecutive steps. First, the generation of HNO_3 is calculated; second, the generation of HCl is calculated; and third, the pH is calculated based on the presence of these acids, boric acid, and TSP. The calculation uses the boron concentration from the RWSP, SITs, BAMT and RCS. The methodology used by the code to calculate the amounts of strong acid generated is consistent with NUREG/CR-5950, "Iodine Evolution and pH Control." Although HI is also a strong acid, relatively small amounts are likely to be in containment. The licensee assumed that 5 percent of the total iodine appears as HI. The licensee did not take credit for the addition of CsOH into containment, which would increase pH, therefore making the calculation more conservative.

The licensee calculated the pH for different time intervals. The staff performed hand calculations to verify the resulting pH value after 30 days. The staff's independent verification demonstrated the containment sump pH would remain above 7 for at least 30 days, consistent with the licensee's submittal.

Based on the review of the licensee's analyses and the staff's independent verifications, the staff concludes the licensee's proposal to be acceptable.

2.2.2 Iodine Spray Removal Coefficients

Radioactive iodine is released to the containment atmosphere in three different forms: elemental, particulate, and organic. For elemental iodine, the licensee considered two distinct mechanisms by which radioactive iodine could be removed: CSs and natural deposition on containment walls. Particulate iodine is only removed by the CSs. There is no effective mechanism for removing organic iodide from the containment atmosphere. Removal of iodine from the containment atmosphere is controlled by two types of parameters: those controlling the rates of removal, called lambdas (λ), and those determining the maximum amount that can be removed, called decontamination factors (DF).

Removal rates of iodine by sprays is a function of volumetric flow of the spray solution, which is reflected in the corresponding spray removal coefficient λ. Although, in some plants these flows may change in time, in Waterford 3 they stay constant; therefore, one value of λ needs to be determined for the whole post-LOCA operation of the plant.

The values of the spray removal coefficient for elemental iodine are calculated by the equation given in Section 6.5.2 of the SRP. However, the SRP sets an upper limit of 20 hr^{-1} on the highest acceptable value. Therefore, regardless of what the calculated value of λ is, it cannot exceed 20 hr^{-1}. Since the licensee's determined λ was considerably higher than 20 hr^{-1}, this value was used in the calculation.

The staff performed independent hand calculations of the λ values for elemental, particulate and natural deposition spray removal coefficients using the equations set forth in Section 6.5.2 of the SRP. The staff concurs with the licensee's results because the values calculated and used in the analyses by the licensee were consistent with the staff's hand calculations. Furthermore, the staff finds the licensee's results to be conservative because they are lower than the maximum limits specified in the SRP.

The DF for iodine in the containment sump is a function of the partition coefficient for iodine, the volume of the containment sump and sump overflow, and the containment building net free volume. The partition coefficient for iodine is a function of the temperature of the sump water, its pH, and its iodine content. According to the SRP, the maximum DF cannot exceed the value of 200 for elemental iodine. The value of the calculated DF for elemental iodine was 28.6. Because the removal mechanisms for particulate and organic iodides are significantly different and slower than that for elemental iodine, there is no need to limit the DF for particulate and organic iodides. The staff performed independent hand calculations of the DF value for the elemental spray removal coefficient using the equation set forth in Section 6.5.2 of the SRP. Even though a DF was calculated by the licensee, since no credit for removal of elemental iodine is being taken, the DF in this case would not be used in the offsite and control room dose calculation.

The staff performed an independent verification of the iodine removal coefficients and DF calculated by the licensee for removal of the elemental iodine from the post-accident containment atmosphere. Based on its results, the staff finds all the values reported by the licensee to be conservative and, therefore, acceptable.

<u>Summary</u>

The staff reviewed the methodology for maintaining the containment sump pH above 7 for the 30-day period after a LOCA. The staff also performed hand calculations to verify the iodine removal coefficients and DF calculated by the licensee for removal of elemental and particulate iodine from the post-accident containment atmosphere.

After an accident, the pH of the containment sump water is determined by the amounts of acidic and basic chemical materials either released from the damaged core or generated in containment and subsequently dissolved in the sump water. It is important to control this pH because, if it falls below 7, radioactive iodine could be released to the containment atmosphere. The addition of a buffering agent such as TSP will keep the water pH above 7, therefore preventing the iodine from being released. The licensee's analysis has indicated that containment sump water will remain greater than 7 for at least 30 days. The staff reviewed the licensee's methodology for determining pH and performed an independent evaluation of the licensee's calculations. Based on its evaluation, the staff concludes that the licensee's proposed actions will maintain the sump water pH greater than 7 for 30 days following a LOCA, thus preventing the release of radioactive iodine into the containment atmosphere.

The staff also evaluated the methods provided by the licensee for the removal of radioactive iodine from the containment atmosphere following a LOCA. The analysis has indicated that all iodine removal coefficients and the decontamination factors are based on the conservative assumptions incorporated in the licensee's methodology. Based on its evaluation, the staff concludes that the methods for removal of radioactive iodine from the post-accident containment atmosphere are acceptable.

2.3 Atmospheric Relative Concentration Estimates

2.3.1 Meteorological Data

The licensee used five years of hourly onsite meteorological data collected during calendar years 1997 through 2001 to generate new EAB, LPZ, and control room air intake χ/Q values. These data were provided for NRC staff review in the form of hourly meteorological data files and joint frequency distributions, input to the ARCON96 and PAVAN atmospheric dispersion computer codes. The resulting χ/Q values represent a change from those values used in the current Waterford 3 UFSAR analyses.

The licensee's data recovery rate for each measured parameter exceeded 95 percent during each of the five years. To assist in obtaining a high recovery rate, data from the primary measurement tower were supplemented by measurements from the back-up tower. The Waterford 3 UFSAR states that the meteorological measurement program is designed to comply with the recommendations of RG 1.23, "Onsite Meteorological Programs."

All releases were considered to be ground level and, as such, were modeled using wind data measured at 33 feet (10 meters (m)) above plant grade. Wind measurements were also made at the 199-foot (61.6-m) level. Stability class was based on delta-temperature measurements made between the 199-foot and 33-foot levels on the onsite meteorological towers.

The staff performed a quality review of the 1997 through 2001 onsite hourly meteorological data using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using computer spreadsheets. With respect to atmospheric stability measurements, stable and neutral conditions were consistently reported to occur at night and unstable and neutral conditions during the day. The frequency, length, and time of occurrence of stable and unstable atmospheric conditions were generally congruent with expected meteorological conditions. Wind speed and direction frequency distributions for each measurement channel were also consistent from year to year and when comparing measurements between the two heights.

A comparison of the joint frequency distribution derived by the staff from the licensee's hourly meteorological database with the joint frequency distribution used by the licensee as input to PAVAN showed reasonably good agreement. The licensee generated its joint frequency distribution as a "representative year" by averaging the yearly joint frequency distributions for the five years rather than generating a joint frequency distribution directly from the hourly data as was done by the staff. A comparison of lower level 1997 through 2001 data with the lower level July 1972 through June 1975 and February 1977 through February 1978 data summaries presented in the Waterford UFSAR also shows reasonable agreement.

In summary, the staff has reviewed the available information relative to the onsite meteorological measurements program, the hourly 1997 through 2001 data, and the PAVAN meteorological data input files provided by the licensee. On the basis of this review, staff concludes that these data provide an acceptable basis for making estimates of atmospheric dispersion estimates for DBA assessments.

2.3.2 EAB/LPZ Atmospheric Dispersion Factors

The licensee calculated EAB and LPZ χ/Q values using the 1997 through 2001 onsite meteorological data and the PAVAN computer code (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Plants"), which implements the guidance provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." The EAB and LPZ distances were input as 914 and 3200 m, respectively, in all 16 downwind sectors. All releases were considered to be ground level. A containment building height of 61 m and a minimum containment building cross-sectional area of 2468 m^2 were used to model building wake effects. The staff qualitatively reviewed the inputs to the PAVAN computer runs and found the inputs to be generally consistent with site configuration drawings and staff practice. The staff also reran the PAVAN code using a joint frequency distribution generated directly from the hourly meteorological data and confirmed that the licensee's χ/Q values appear to be reasonable.

In summary, the staff has reviewed the licensee's assessments of EAB and LPZ post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric diffusion modeling. The resulting EAB and LPZ χ/Q values are presented in Table 13. On the basis of this review, the staff concludes that these EAB and LPZ χ/Q values are acceptable for use in DBA EAB and LPZ dose assessments.

2.3.3 Control Room Atmospheric Dispersion Factors

The licensee calculated control room air intake χ/Q values using 1997 through 2001 onsite meteorological data, the ARCON96 atmospheric computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"), and RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." Values were calculated for releases to both the east and west control room air intakes from the following eleven postulated release locations: east and west side MSSVs, ADVs, and MSL; the plant stack; the FHB, truck bay, and personnel doors; and the containment equipment hatch and purge intake. The staff qualitatively reviewed the inputs to the ARCON96 computer runs and found them generally consistent with site configuration drawings and staff practice. Specific areas of note are as follows:

- Postulated releases were modeled as ground level point sources assuming a straight line flow path from the release point to the receptor, ignoring intervening structures, thus underestimating the true travel distance. Source-receptor directions input into ARCON96 were based upon true north and release heights above grade were used as inputs, assuming no vent flow. The staff performed checks of the inputs and found the assumptions and inputs acceptable.
- The east ADV is less than 10 ms from the east intake. RG 1.194 states that such situations should be addressed on a case-by-case basis. Therefore, the licensee performed sensitivity comparisons of resultant χ/Q values at a range of distances using ARCON96, estimated plume rise (for which they had not taken credit in their estimates) using RG 1.194, estimated mixing of the effluent with steam and air prior to release from the ADV, and provided a description of several other conservatisms

in the dose assessment. The licensee noted that the volumetric flow rate from the ADV at the time of lowest flow, by itself, indicated effluent dilution of the same magnitude as the χ/Q value calculated for the 0 to 2 hour time period using the ARCON96 computer code. The staff reviewed the information provided by the licensee and finds the magnitude of the χ/Q values for the east ADV to the east control room air intake acceptable for this specific case. However, the staff does not approve the use of the ARCON96 code at distances less than about 10 ms without adequate case-specific justification.

In summary, the staff has reviewed the licensee's assessment of control room post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. On the basis of this review, the staff concludes that the control room χ/Q values for the postulated source/receptor pairs listed below in Table 15 are acceptable.

2.3.4 Choice of Favorable Control Room Intake

In calculating the dose in the control room for each of the analyzed accidents, the licensee took credit for the control room operators choosing the control room ventilation pressurization air intake with the lesser amount of measured radioactivity. The licensee's analyses assume this occurs at 2 hours into the event, with the exception of the FHA, which assumed no credit for the more favorable intake. The licensee takes credit for the pressurization intake of relatively clean air by reducing the calculated control room χ/Qs, listed below in Table 15, by a factor of four for the time period after the favorable intake has been selected and the contaminated intake isolated. SRP 6.4, "Control Room Habitability System," and RG 1.194 both give guidance on this analysis credit for plants that have two outside air intakes that are designed and located with the intent of providing a low contamination intake regardless of wind direction. The guidance states that for the credit of a reduction of a factor of four the following should apply: (1) the two intakes should each meet applicable design criteria of an ESF; (2) the intakes are not within the same wind direction window; (3) there are redundant ESF-grade radiation monitors within each intake with control room indication and alarm to monitor the intakes; and (4) requisite steps to select the least contaminated outside air intake and provisions for monitoring to ensure the favorable intake is in use throughout the event are addressed in procedures and operator training. The staff asked the licensee to verify all these conditions are met, and the licensee provided this verification in its supplemental letter dated October 13, 2004. The control room χ/Qs used by the licensee in its dose analyses and listed in its supplements dated July 15, August 19, and October 19, 2004, include a reduction by a factor of four in the pressurization intake χ/Q for the time periods after four hours, for those accidents that assume the operators choose the favorable intake at that time. The staff finds that the licensee applied this credit for manually choosing the least contaminated control room air intake correctly.

2.4 Summary Alternative Source Term 10 CFR 50.67

Implementation of the AST will not increase the quantities or alter the types of radioactive material actually released if an event were to occur. Implementation of the AST also has no effect on the actual or calculated effluents arising from normal operation. With respect to occupational doses, the AST is, again, only a change in dose calculation inputs and methodology. Calculated doses meet TEDE criteria. No aspect of implementing the AST involves facility equipment, procedure, or process changes that would increase actual onsite doses if an event were to occur. The AST does not result in actual or calculated changes in the normal radiation levels in the facility, or in the type or quantity of radioactive materials processed during normal operation. The NRC staff has determined that the amendments involve 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.

As described in this SE, the NRC staff reviewed the assumptions, inputs, and methods used by Entergy to assess the radiological impacts of implementing a full-scope AST and EPU at Waterford 3. The NRC staff finds that Entergy used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The NRC staff compared the doses estimated by Entergy to the applicable criteria identified in Section 2.0. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses will continue to comply with these criteria, for DBAs at Waterford 3.

This licensing action is considered to be a full implementation of the AST. With this approval, the previous accident source term in the Waterford 3 design basis is superseded by the AST proposed by Entergy. The previous offsite and control room accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67 or fractions thereof, as defined in RG 1.183. Future radiological analyses performed to demonstrate compliance with regulatory requirements must address all characteristics of the AST and the TEDE criteria as described in the Waterford 3 design basis.

Table 1

Licensee Calculated DBA Radiological Consequences for AST and EPU

Accident	TEDE (rem) at location		
	EAB	LPZ	control room
MSLB inside containment, 2% fuel melting	0.60	0.19	4.89
MSLB outside containment/FWLB			
Pre-existing iodine spike	0.22	0.08	2.20
Accident induced iodine spike	0.23	0.12	3.62
LRA, 15% fuel failure	0.55	0.21	3.19
REA, 15% fuel failure	1.03	0.65	3.19
Letdown Line Break			
Pre-existing iodine spike	4.22	0.67	1.00
Accident induced iodine spike	0.63	0.10	0.14
SGTR			
Pre-existing iodine spike	0.99	0.21	4.85
Accident induced iodine spike	0.44	0.09	2.56
LBLOCA	5.30	2.37	1.47
SBLOCA	1.96	1.08	3.93
FHA	0.55	0.09	0.19
IADVO event			
Pre-existing iodine spike	0.22	0.08	2.20
Accident induced iodine spike	0.23	0.12	3.62
Excess Load, 8% fuel failure	0.33	0.12	1.87

Table 2

Analysis Assumptions for MSLB Inside Containment

Parameter	Value
Power level, MWt	3735
Core fuel rods melted, %	2
Radial peaking factor	1.65
Core inventory in licensee's submittal (7/15/04)	Table 1-1
Fraction of core activity in failed fuel rod gap	
Iodines and noble gases	0.10
Alkali metals (Cs & Rb-86)	0.12
Containment Leakage Pathway	
Containment leak rate, % vol/day	
0-24 hours	0.50
1-30 days	0.25
Natural deposition	Not credited
Spray fission product removal	Not credited
Iodine chemical form, %	
Elemental	4.85
Organic	0.15
Particulate	95
Secondary Steaming Pathway	
Steam generator in faulted loop	
Initial water mass, lbm	241,450
Primary to secondary leak rate, gpd	540
Iodine partition factor	1
Steam generator in intact loop	
Initial water mass, lbm	241,450
Primary to secondary leak rate, gpd	150
Iodine partition factor	
0-30 min	1
> 30 min	100
Primary coolant liquid mass not including pressurizer, lbm	395,502
Duration of release, hr	7.5
Steam release through unaffected SG, lbm	
0-2 hr	627,512
2-8 hr	858,838
Atmospheric dispersion values, offsite receptors	Table 13
Control room parameters	Table 14
Control room X/Qs listed in licensee's submittal (7/15/04)	Table 7-1

Table 3

Analysis Assumptions for MSLB Outside Containment/Feedwater Line Break

Parameter	**Value**
Power level, MWt	3735
Accident induced iodine spike	
Primary coolant iodine concentration, μCi/gm DEI-131	1
Accident induced iodine spiking factor	500
Pre-existing iodine spike	
Primary coolant iodine concentration, μCi/gm DEI-131	60
Iodine chemical form, %	
Elemental	97
Organic	3
Primary coolant mass w/o pressurizer, lbm	395,502
Secondary coolant mass, lbm	482,900
Primary to secondary leak rate, gpd	
Affected SG	540
Unaffected SG	150
SG steaming partition factor	
0-4 hr	1
> 4 hr	100
Duration of release, hr	36
Steam release through SG, lbm	
0-2 hr	588,365
2-8 hr	1,333,286
Atmospheric dispersion values, offsite receptors	Table 13
Control room parameters	Table 14
Control room X/Qs listed in licensee's submittal (7/15/04)	Table 9-1

Table 4

Analysis Assumptions for LRA

Parameter	Value
Power level, MWt	3735
Core fuel rods failed, %	15
Radial peaking factor	1.65
Fraction of core activity in failed fuel rod gap	
Iodines and noble gases	0.10
Alkali metals (Cs & Rb-86)	0.12
Primary coolant mass w/o pressurizer, lbm	395,502
Total secondary coolant mass, lbm	482,900
Primary to secondary leak rate, gpd per SG	150
SG steaming partition factor	
0-30 min	10
> 30 min	100
Duration of release, hr	7.5
Steam release through SGs, lbm	
0-2 hr	609,744
2-8 hr	858,838
Atmospheric dispersion values, offsite receptors	Table 13
Control room parameters	Table 14
Control room X/Qs listed in licensee's supplement (8/19/04)	Table 4-1

Table 5

Analysis Assumptions for REA

Parameter	Value
Power level, MWt	3735
Core fuel rods failed, %	15
Radial peaking factor	1.65
Fraction of fission-product inventory released to coolant from failed fuel rods	
Iodines and noble gases	0.10
Alkali metals (Cs & Rb-86)	0.12
Primary coolant mass w/o pressurizer, lbm	395,502
Secondary Steaming Pathway	
Primary to secondary leak, gpd per SG	150
Total secondary coolant mass, lbm	482,900
SG steaming partition factor	
0-30 min	10
> 30 min	100
Iodine chemical form, %	
Elemental	97
Organic	3
Duration of release, hr	7.5
Steam release through SGs, lbm	
0-2 hr	609,744
2-8 hr	858,838
Containment Leakage Pathway	
Containment leak rate, volume % per day	
0 to 24 hr	0.5
>24 hr	0.25
Iodine chemical form, %	
Elemental	4.85
Organic	0.15
Particulate	95
Atmospheric dispersion values, offsite receptors	Table 13
Control room parameters	Table 14
Control room X/Qs listed in licensee's submittal (7/15/04)	Table 10-1

Table 6

Analysis Assumptions for Letdown Line Break Accident

Parameter	Value
Power level, MWt	3735
Accident induced iodine spike	
Primary coolant iodine concentration, μCi/gm DEI-131	1
Accident induced iodine spiking factor	500
Pre-existing iodine spike	
Primary coolant iodine concentration, μCi/gm DEI-131	60
Iodine chemical form, %	
Elemental	97
Organic	3
Primary coolant mass w/o pressurizer, lbm	395,502
Secondary coolant mass, lbm	482,900
Primary to secondary leak rate, gpd per SG	150
SG steaming partition factor	
0-30 min	10
> 30 min	100
Duration of release, hr	8
Break release, lbm	
0-30 min	78,000
Steam release through SGs, lbm	
0-2 hr	794,217
2-8 hr	1,357,617
Atmospheric dispersion values, offsite receptors	Table 13
Control room parameters	Table 14
Control room X/Qs listed in licensee's supplement (8/19/04)	Table 7-1

Table 7

Analysis Assumptions for Steam Generator Tube Rupture Accident

Parameter	Value
Power level, MWt	3735
Accident induced iodine spike	
Primary coolant iodine concentration, μCi/gm DEI-131	1
Accident induced iodine spiking factor	335
Pre-existing iodine spike	
Primary coolant iodine concentration, μCi/gm DEI-131	60
Iodine chemical form, %	
Elemental	97
Organic	3
Primary coolant mass, lbm	467,000
Secondary coolant mass, lbm	106,300
Primary to secondary leak rate, gpd	
Intact SG	540
SG steaming partition factor	100
Duration of release, hr	8
Break release to ruptured SG, lbm	325,700
Steam release through intact SG, lbm	
0-2 hr	351,400
0-8 hr	910,100
Steam release through ruptured SG, lbm	
Before isolation (1980 sec)	139,000
0-8 hr	245,600
Atmospheric dispersion values, offsite receptors	Table 13
Control room parameters	Table 14
Control room X/Qs listed in licensee's supplement (10/19/04)	Table 5-1

Table 8

Analysis Assumptions for Large Break Loss of Coolant Accident

Parameter	Value
Power level, MWt	3735
Source term model	RG 1.183
Primary containment free volume, ft^3	2,568,000
Primary containment leakage, volume % per day	
0 to 24 hours	0.50
>24 hours	0.25
Containment leakage filtered by CVAS, percent	54
CVAS iodine filtration efficiency all species, percent	99
Containment leakage filtered by SBVS, percent	40
SBVS iodine filtration efficiency all species, percent	90
Sprayed volume of containment, %	80
Maximum spray delay time, sec	60
Spray iodine removal coefficient, hr^{-1}	
Elemental	0
Organic	0
Particulate/Aerosol	
Until PF = 50	3.596
PF > 50	0.3596
Mixing rate between sprayed and unsprayed regions, cfm	17,122
Iodine natural deposition, hr^{-1}	
Elemental	0.4
Organic	0
Particulate/Aerosol	Powers 10% Factor
ECCS leakage, gpm	0.5
ECCS leakage flashing fraction	0.10
ECCS recirculation start, min	23.4
Atmospheric dispersion values, offsite receptors	Table 13
Control room parameters	Table 14
Control room X/Qs listed in licensee's supplement (10/19/04)	Table 4-1

Table 9

Analysis Assumptions for Small Break Loss of Coolant Accident

Parameter	Value
Power level, MWt	3735
Core fuel rods failed, %	100
Fraction of core activity in failed fuel rod gap	
Iodines and noble gases	0.05
Alkali metals (Cs & Rb-86)	0.05
Primary coolant mass w/o pressurizer, lbm	395,502
Total secondary coolant mass, lbm	482,900
Primary to secondary leak rate, gpd per SG	75
SG steaming partition factor	
0-30 min	10
> 30 min	100
Duration of release, hr	7.5
Steam release through SGs, lbm	
0-2 hr	627,512
2-8 hr	858,838
Primary containment free volume, ft^3	2,568,000
Primary containment leakage, volume % per day	
0 to 24 hours	0.50
>24 hours	0.25
Iodine natural deposition, hr^{-1}	
Elemental	0.4
Organic	0
Particulate/Aerosol	Powers 10% Factor
Iodine spray removal	Not credited
Atmospheric dispersion values, offsite receptors	Table 13
Control room parameters	Table 14
Control room X/Qs listed in licensee's submittal (7/15/04)	Table 6-1

Table 10

Analysis Assumptions for Fuel Handling Accidents

Parameter	Value
Power level, MWt	3735
Peaking factor	1.65
Number of fuel rods in core	51212
Number of fuel rods damaged	60
Reactor shutdown time before fuel movement, hr	72
Core fractions released from damaged rods	
I-131	0.12
Other halogens	0.05
Kr-85	0.14
Other noble gases	0.05
Alkali metals	0.12
Iodine effective pool decontamination factor	200
Duration of release	Instantaneous
Atmospheric dispersion values, offsite receptors	Table 13
Control room parameters	Table 14
Control room X/Qs listed in licensee's submittal (7/15/04)	Table 11-1

Table 11

Analysis Assumptions for Inadvertent ADV Opening Event

Parameter	Value
Power level, MWt	3735
Accident induced iodine spike	
Primary coolant iodine concentration, µCi/gm DEI-131	1
Accident induced iodine spiking factor	500
Pre-existing iodine spike	
Primary coolant iodine concentration, µCi/gm DEI-131	60
Iodine chemical form, %	
Elemental	97
Organic	3
Primary coolant mass w/o pressurizer, lbm	395,502
Total secondary coolant mass, lbm	482,900
Primary to secondary leak rate, gpd	
Unaffected SG	150
Affected SG	540
SG steaming partition factor	
0-4 hr	1
> 4 hr	100
Duration of release, hr	36
Steam release through SGs, lbm	
0-2 hr	588,365
2-8 hr	1,333,286
Atmospheric dispersion values, offsite receptors	Table 13
Control room parameters	Table 14
Control room X/Qs listed in licensee's supplement (8/19/04)	Table 5-1

Table 12

Analysis Assumptions for Excess Load Event

Parameter	Value
Power level, MWt	3735
Core fuel rods failed, %	8
Radial peaking factor	1.65
Fraction of core activity in failed fuel rod gap	
Iodines and noble gases	0.10
Alkali metals (Cs & Rb-86)	0.12
Primary coolant mass w/o pressurizer, lbm	395,502
Total secondary coolant mass, lbm	482,900
Primary to secondary leak rate, gpd per SG	150
SG steaming partition factor	
0-30 min	10
> 30 min	100
Duration of release, hr	7.5
Steam release through SGs, lbm	
0-2 hr	609,744
2-8 hr	858,838
Atmospheric dispersion values, offsite receptors	Table 13
Control room parameters	Table 14
Control room X/Qs listed in licensee's supplement (8/19/04)	Table 6-1

Table 13

Waterford EAB and LPZ Atmospheric Dispersion Factors

Time (hr)	Receptor Location	X/Q (sec/m^3)
0 - 2 hours	Exclusion Area Boundary	4.31 E-04
0 - 8 hours	Low Population Zone	6.58 E-05
8 - 24 hours	Low Population Zone	4.45 E-05
1 - 4 days	Low Population Zone	1.91 E-05
4 - 30 days	Low Population Zone	5.88 E-06

Table 14

Control Room Analysis Assumptions

Parameter	**Value**
Control room volume, ft^3	220,000
Recirculation flow rate, cfm	3,800
Filter efficiency, %	
Elemental	99
Organic	99
Particulate	99
Pressurization flow rate, cfm	
Maximum	225
Minimum after 2 hours	50
Unfiltered inleakage assumption, cfm	
MSLB, inside containment	100
MSLB, outside containment	100
LRA	150
REA	150
Letdown line break	150
SGTR	100
LBLOCA	100
SBLOCA	100 (0-2 hr), 65 (>2 hr)
FHA	200
IADVO event	100
Excess load	150
Control room occupancy factors	
0-24 hr	1.00
1-4 days	0.60
4-30 days	0.40

Table 15

Waterford Control Room (CR) Atmospheric Dispersion Factors (X/Q) for DBA Assessment X/Q Values (sec/m^3)

	East MSSV		West MSSV		East ADV	
Time Period	East CR Air Intake	West CR Air Intake	East CR Air Intake	West CR Air Intake	East CR Air Intake	West CR Air Intake
0 to 2 hours	4.36 E-02	1.37 E-03	1.52 E-03	7.40 E-03	1.06 E-01	1.23 E-03
2 to 8 hours	3.08 E-02	9.34 E-04	9.44 E-04	5.44 E-03	7.45 E-02	8.31 E-04
8 to 24 hours	1.33 E-02	4.48 E-04	4.00 E-04	2.46 E-03	3.30 E-02	4.00 E-04
1 to 4 days	9.01 E-03	2.99 E-04	2.81 E-04	1.92 E-03	2.31 E-02	2.63 E-04
4 to 30 days	6.57 E-03	2.10 E-04	2.07 E-04	1.50 E-03	1.62 E-02	1.85 E-04

	West ADV		East MSL		West MSL	
Time Period	East CR Air Intake	West CR Air Intake	East CR Air Intake	West CR Air Intake	East CR Air Intake	West CR Air Intake
0 to 2 hours	1.36 E-03	7.50 E-03	5.09 E-02	1.44 E-03	1.54 E-03	9.28 E-03
2 to 8 hours	8.29 E-04	5.62 E-03	3.26 E-02	9.78 E-04	9.62 E-04	6.84 E-03
8 to 24 hours	3.55 E-04	2.57 E-03	1.39 E-02	4.68 E-04	4.11 E-04	3.11 E-03
1 to 4 days	2.48 E-04	2.04 E-03	8.81 E-03	3.05 E-04	2.89 E-04	2.37 E-03
4 to 30 days	1.85 E-04	1.57 E-03	6.87 E-03	2.18 E-04	2.15 E-04	1.85 E-03

Table 15 (cont.)

Waterford Control Room (CR) Atmospheric Dispersion Factors (X/Q) for Design Basis Accident Assessment X/Q Values (sec/m^3)

	Plant Stack		FHB Truck Bay		FHB Personnel Door	
Time Period	**East CR Air Intake**	**West CR Air Intake**	**East CR Air Intake**	**West CR Air Intake**	**East CR Air Intake**	**West CR Air Intake**
0 to 2 hours	2.77 E-03	2.06 E-03	7.50 E-04	7.63 E-04	9.75 E-04	1.05 E-03
2 to 8 hours	1.78 E-03	1.56 E-03	6.15 E-04	6.32 E-04	7.74 E-04	8.72 E-04
8 to 24 hours	7.22 E-04	7.16 E-04	2.62 E-04	2.95 E-04	3.33E-04	4.02 E-04
1 to 4 days	5.27 E-04	5.49 E-04	1.82 E-04	2.27 E-04	2.22 E-04	3.08 E-04
4 to 30 days	4.05 E-04	4.32 E-04	1.25 E-04	1.70 E-04	1.55 E-04	2.29 E-04

	Containment Hatch		Containment Purge	
Time Period	**East CR Air Intake**	**West CR Air Intake**	**East CR Air Intake**	**West CR Air Intake**
0 to 2 hours	1.22 E-03	1.93 E-03	1.55 E-02	1.68 E-03
2 to 8 hours	8.54 E-04	1.60 E-03	1.01 E-02	1.20 E-03
8 to 24 hours	3.64 E-04	7.42 E-04	4.18 E-03	5.75E-04
1 to 4 days	2.43 E-04	5.61 E-04	2.72 E-03	3.90 E-04
4 to 30 days	1.86 E-04	4.24 E-04	2.13 E-03	2.67 E-04

3.0 COMMITMENTS

The licensee included regulatory commitments in its application and supplements.

The commitment relevant to the NRC staff evaluation is as follows:

> Waterford 3 plant procedures will be revised to specify a maximum ESF leakage of half the value specified in LOCA radiological analyses, consistent with RG 1.183.
>
> Type: One Time Action
> Scheduled Completion Date (if required): EPU/AST Implementation

(Reference: Supplement dated September 1, 2004, ADAMS Accession No. ML042470194)

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the regulatory commitments are provided by the licensee's administrative processes, including its commitment management program. (See Regulatory Issue Summary 2000-17, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff," dated September 21, 2000). The staff has determined that the above commitment does not warrant the creation of regulatory requirements which would require prior NRC approval of subsequent changes.

4.0 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 comments.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes the surveillance requirements. The 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (69 FR 51488, dated August 19, 2004). Accordingly, the amendments meet 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 amendments.

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

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Date: March 29, 2005

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