Mr. Joseph E. Venable January 8, 2003
Vice President Operations
Entergy Operations, Inc.
17265 River Road
Killona, LA 70066-0751

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT RE: LETDOWN LINE BREAK DOSE CONSEQUENCES REVISION (TAC NO. MB3231)

Dear Mr. Herron:

The Commission has issued the enclosed Amendment No. 184 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 15, 2001, as supplemented by letter dated August 27, 2002.

The amendment provides additional information to support a modification to TS 3.4.7 and limits Reactor Coolant System activity permitted by the ACTION statement to 60 microcuries per gram (µCi/gm) at all power levels. The letdown line break accident analysis in the Final Safety Analysis Report was also changed.

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

Sincerely,

/RA/

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

Docket No. 50-382

Enclosures: 1. Amendment No.184 to NPF-38
2. Safety Evaluation

cc w/encls: See next page

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ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 184
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 October 15, 2001, as supplemented by letter dated August 27, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   2. Technical Specifications and Environmental Protection Plan

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 184, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

   Additionally, the license is amended to approve changes to the Final Safety Analysis Report (FSAR) Section 15.6 and addition of Figure 15.1.75a. The changes concern the new analyses of the letdown break dose consequences as set forth in the application for amendment by EOI, dated October 15, 2001, as supplemented by letter dated August 27, 2002. EOI shall update the FSAR to reflect the revised licensing basis authorized by this amendment in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance and in the next periodic update to the FSAR in accordance with 10 CFR 50.71(e). With regards to the FSAR, implementation of the amendment is the incorporation into the FSAR of the changes to the description of the facility as described in the licensee's application dated October 15, 2001, as supplemented by letter dated August 27, 2002, and evaluated in the staff's Safety Evaluation attached to this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment: Changes to the Technical Specifications

Date of Issuance: January 8, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 184

TO FACILITY OPERATING LICENSE NO. NPF-38

DOCKET NO. 50-382

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

Remove	Insert
XIX	XIX
XXI	XXI
3/4 4-24	3/4 4-24
3/4 4-25	3/4 4-25
3/4 4-26	3/4 4-26
3/4 4-27	Deleted

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 184 TO

FACILITY OPERATING LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By application dated October 15, 2001 (Reference 1), as supplemented by letter dated August 27, 2002 (Reference 2), Entergy Operations, Inc. (Entergy, the licensee), submitted a request for changes to the Waterford Steam Electric Station, Unit 3 (Waterford 3), Technical Specifications (TSs). The requested changes provide additional information to support a modification to TS 3.4.7 and limits the permitted 48-hour Reactor Coolant System (RCS) activity to 60 microcuries per gram (μCi/gm) Dose Equivalent I-131 (DEI-131) at all reactor power levels, replacing the previous power dependent curve. The letdown line break accident analysis in the Final Safety Analysis Report (FSAR) was also changed.

In Reference 1, the licensee assumed an initial condition of three charging pumps in operation to assure that the most severe radioactive releases would be considered. Because of this assumption, the Standard Review Plan (SRP) acceptance criterion of a small fraction (10 percent) of the 10 CFR Part 100 limits was exceeded and Nuclear Regulatory Commission (NRC) approval was required in accordance with 10 CFR 50.59. This application was noticed in the Federal Register on November 28, 2001 (66 FR 59504). After discussions with NRC, the licensee, in Reference 2, assumed an initial condition of a single operating charging pump which provides a suitable licensing basis and has sufficient conservatism to accommodate two- and three-pump operating scenarios that may exist during the operating cycle. The results of this analysis fall within the acceptance criteria contained in the SRP, but the increase in dose exceeds 10 percent of the difference between the currently approved dose and the regulatory limit. Reference 2 was noticed in the Federal Register on October 29, 2002 (67 FR 66009), and superceded the biweekly Federal Register notice dated November 28, 2001 (66 FR 59504).

2.0 BACKGROUND

In a 1998 condition report, Entergy identified a discrepancy between the assumed isolation time of five seconds in the letdown line break analysis and the Technical Requirements Manual (TRM) provision for a 10-second closure time for the isolation valve. During the licensee's reanalysis to account for the increased isolation valve closure time, two additional issues were identified and were documented in another condition report in 1999. The first issue was that the licensee's original letdown line break dose consequences analysis did not include a scenario in which a high RCS activity, equal to the TS 48-hour maximum allowable iodine activity, is assumed to exist at the time of the initiation of the letdown line break. The second issue was that, for the accident induced iodine spiking scenario, the licensee determined that the largest mass and activity release was for a break size that did not result in a reactor trip and isolation until manual actuation 30 minutes after the line break, as opposed to the previous assumption of an automatic reactor trip and isolation at about 426 seconds. The licensee's submittal addressed these issues by making the needed revisions to the letdown line break dose analysis.

The staff finds that the licensee identified the applicable regulatory requirements in section 1.0 of Reference 1. The regulatory requirements on which the staff based its review of the application are 10 CFR 100.11 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC)-19. Staff referred to guidance on review of the design basis accident (DBA) letdown line break dose consequences analysis from NUREG-0800, SRP Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment." This section denotes regulatory acceptance criteria for the letdown line break dose consequences analysis that are a small fraction (i.e., 10 percent) of the Part 100 guideline values for an assumed TS equilibrium iodine concentration in the reactor coolant, with an accident induced spike of 500 times the equilibrium iodine appearance rate. GDC-19 is the applicable requirement for the control room with regard to the dose consequences of DBAs.

3.0 EVALUATION

3.1 Letdown Line Break Dose Consequences Analysis

3.1.1 Background

Entergy evaluated the impact of the proposed changes in methodology to show that applicable regulatory acceptance criteria would continue to be satisfied. The discussion below identifies the inputs and assumptions provided by Entergy and utilized by the staff to perform independent calculations. The results of the staff's independent calculations and evaluations were used to determine the acceptability of the licensee's analysis methodology. The design inputs utilized by the staff to evaluate these accidents and the licensee's calculated results are given in Tables 1 and 2. Inputs and model development issues that warrant further discussion are given below.

The licensee's proposed change to TS Action Statement 3.4.7.a to limit the RCS activity to 60 µCi/gm DEI-131 at all reactor power levels required a revision to the letdown line break dose consequences analysis. This is because the previous letdown line break analysis was not performed at the action statement limit. The other Waterford 3 DBA dose consequences analyses that have only RCS activity as a source term are the main steam line break (MSLB) and steam generator tube rupture (SGTR) accidents. The MSLB and SGTR analyses are currently documented in the Waterford 3 FSAR as having assumed a pre-existing iodine spike of 60 µCi/gm DEI-131 in the RCS. Therefore, these analyses did not have to be revised for the proposed change to TS 3.4.7.a.

3.1.2 Revisions to Letdown Line Break Dose Consequences Analysis

The letdown line break dose consequences analysis assumes a break in the letdown line that releases reactor coolant outside of the reactor containment. The break flow is terminated by isolating the letdown line inside containment. The licensee has recalculated the duration of the spill of coolant from the letdown line and the subsequent mass of coolant released through the break. The review of the letdown line break analysis was based on the assumption that the licensee's transient mass release analysis is first found to be acceptable. All other analysis assumptions not discussed in this safety evaluation were the same as previously documented in the Waterford 3 FSAR.

The staff finds that the licensee generally followed SRP guidance in the modeling of the letdown line break as follows. The licensee evaluated three scenarios for the activity in the reactor coolant available to be released through a break in a letdown line: equilibrium activity, accident induced iodine spiking, and pre-existing iodine spike. The only difference in the analysis inputs among the scenarios was the RCS activity; the rest of the model did not differ. For all the scenarios, the licensee elected to use thyroid dose conversion factors derived from International Commission on Radiological Protection Publication 30 (ICRP-30) in place of those previously used by the licensee. The staff has generically found acceptable use of thyroid dose conversion factors derived from ICRP-30, noted in RIS 2001-19.

3.1.2.1 Equilibrium RCS Activity

In the analysis of this accident scenario, the licensee assumed the RCS activity was at the TS 3.4.7 limit of 1.0 µCi/gm DEI-131, in place of the previous assumption of activity equal to that from failed fuel equaling 1 percent of the core. Though this is a reduction in source term, the staff finds it acceptable because the Waterford 3 TSs require the RCS activity to remain below 1.0 µCi/gm DEI-131 for normal operations and there is no failed fuel expected as result of the letdown line break. The staff's independent calculation confirmed the licensee's results. The licensee's calculated results are listed in Table 2 and are a small fraction of 10 CFR Part 100 dose limits.

3.1.2.2 Accident Induced Iodine Spiking

During its re-analysis of the scenario that assumes an accident induced iodine activity appearance rate spike in the RCS coolant, the licensee determined that the pre-accident letdown flow assumed in the development of FSAR Figure 15.1-75 was non-conservative. The original curve assumed a letdown flow of 44 gallons per minute (gpm), which is approximately equal to the flow with one charging pump in operation. Because site off-normal procedures direct letdown flow to be maximized for RCS cleanup during periods of elevated RCS activity levels, two or possibly three charging pumps may be in operation in the event of high RCS activity. The licensee determined that this discrepancy between the assumption of flow through one charging pump, used in the development of the iodine spiking rate, and the actual expected operation at high RCS activity levels was non-conservative. The dose results of the licensee's reanalysis with the assumption of letdown flow equivalent to three charging pumps in operation (144 gpm) did not meet the SRP 15.6.2 acceptance criteria that the offsite doses are a small fraction of 10 CFR Part 100 limits (10 percent of Part 100 or 30 roentgen equivalent man (rem) to the thyroid). The licensee proposed that, due to the low occurrence probability of the event, plus the conservative nature of the calculation, it should be acceptable to allow the offsite consequences of the letdown line break assuming accident induced iodine spiking to fall well within (25 percent of) 10 CFR Part 100 limits or 75 rem to the thyroid.

The guidance in SRP 15.6.2 gives conservative assumptions that are acceptable to the staff. One scenario assumed iodine spiking as a result of the reactor shutdown or depressurization of the primary system. The spike is modeled by increasing by a factor of 500 the fission product inventory release rate from the fuel that is associated with the TS equilibrium activity level. The fission product of interest is iodine, and this release rate from the fuel is also referred to as the iodine appearance rate. For the accident induced spiking scenario, the TS equilibrium iodine activity is present in the RCS liquid at the beginning of the accident and the iodine activity due to the increased iodine appearance rate is added over the duration of the spike. The equilibrium iodine appearance rate is calculated by a mass balance equation, with one of the iodine removal variables being the letdown rate.

Because Waterford 3 has one charging pump in operation to maintain the RCS activity level during normal operations, the iodine appearance rate is better determined based on one-pump operation. When the letdown rate is increased by assuming more than one pump is running, the iodine appearance rate also increases, so that the total release of iodine activity is higher. Because during normal operation, Waterford 3 has only one pump in operation for letdown and the equilibrium iodine appearance rate is determined for normal operation, the staff's position is that the assumption of three pumps in operation for letdown is overly conservative for the determination of the equilibrium iodine appearance rate at Waterford 3. The staff communicated this position to the licensee in a telephone conference call on May 10, 2002, and by letter dated June 20, 2002, which was a request for additional information on the subject amendment request. During this conference call, the staff also informed the licensee that the staff was not apt to accept the dose consequences of the accident induced spiking not meeting the SRP acceptance criteria, considering the overly conservative nature of the licensee's spiking model. Subsequently, the licensee performed the accident induced iodine spiking scenario using the assumption of one-pump letdown in the determination of the iodine appearance rate, with results that are within a small fraction of 10 CFR Part 100 limits. The staff's independent calculation confirmed the licensee's results. The licensee's calculated results are listed in Table 2.

3.1.2.3 Pre-existing Iodine Spike

In the analysis of this accident scenario, the licensee assumed the RCS activity was at the TS Action Statement 3.4.7.a 48-hour maximum of 60 µCi/gm DEI-131. The staff's independent calculation confirmed the licensee's results. The licensee's calculated results are listed in Table 2 and are within 10 CFR Part 100 dose limits.

3.1.3 Control Room Habitability

Entergy did not perform an analysis of the dose consequences in the control room as a result of the letdown line break. By Reference 2, in response to the staff's question why the calculation was not done, the licensee asserted that the current FSAR control room habitability analysis, based on the loss-of-coolant accident (LOCA), bounds the revised letdown line break accident. The licensee based this assertion on the following two determinations: 1) the assumed activity release is larger for the LOCA, and 2) because of the locations of the LOCA release and the letdown line break release, the atmospheric dispersion factors used to determine the control room dose for the LOCA would bound those for the letdown line break. Since the activity release and the atmospheric dispersion factors are both directly proportional to the dose results, the LOCA control room dose would bound that for the letdown line break. Based on the preceding discussion, the staff finds that there is reasonable assurance that the doses in the control room will continue to meet the requirements of GDC-19 for the letdown line break.

The staff is currently working toward resolution of generic issues related to control room habitability, in particular, the validity of control room infiltration rates assumed by licensees in analyses of control room habitability. About 30 percent of power plant control rooms have been tested using enhanced test methods. In all but one case, the measured infiltration rates exceeded the values assumed in the design basis analyses. While in each case the affected licensee was able to either reduce the excessive infiltration or show the acceptability of the observed infiltration, the collective experience has caused concerns regarding those facilities that have not performed the enhanced testing. The staff is currently working to resolve these concerns. The staff has determined that there is reasonable assurance that the Waterford 3 control room will be habitable during DBAs and that this amendment may be approved before the resolution of this generic issue. The approval of this amendment does not relieve Entergy of any actions that may be necessary in the future as this generic issue is resolved.

3.1.4 Summary

Based on the considerations discussed above and the information provided by Entergy regarding the letdown line break, the staff finds that there is reasonable assurance that the postulated dose consequences of the design basis letdown line break will be less than the dose criteria of 10 CFR Part 50, Appendix A, GDC-19 and Section 6.4 of NUREG-0800. The staff also finds that there is reasonable assurance that the postulated dose consequences of the design basis letdown line break are within the guidelines of 10 CFR Part 100 and meet the acceptance criteria of Section 15.6.2 of NUREG-0800. Therefore, the staff finds that the proposed changes to the Waterford 3 TSs are acceptable with regard to the dose consequences of DBAs.

3.2 Letdown Line Break Mass Release Consequences Analysis

The purpose of this review is to confirm that the licensee's reanalysis is based on acceptable methods, the assumptions used in the reanalysis are adequate, and the results of the analysis are acceptable. The staff has reviewed the licensee's results of the letdown line break reanalysis for the RCS mass release determination.

3.2.1 Computer Code Used for the RCS Mass Release Calculation

The original analysis was performed with the CEFLASH-4AS computer code. The CEFLASH-4AS code was previously approved by the NRC for use in calculating the hydraulic response of small-break LOCAs. The licensee indicated that CEFLASH-4AS was not capable of modeling a pressurizer pressure control system or a pressurizer level control system. The lack of pressure control models causes a more rapid depressurization and, consequently, an earlier low pressurizer pressure trip. The short transient duration results in less RCS mass release during a transient.

The reanalysis was performed with the CESEC-III code. CESEC-III is also an NRC approved code for use in calculating such system parameters as core power, RCS flow, pressure, temperature, and valve actions during a design-basis transient. It models the pressurizer control systems in automatic mode to delay the time of reactor trip and results in an increased RCS mass release.

The licensee indicated that the original analysis using CEFLASH-4AS did not model the flow resistence in the letdown line between the RCS and outside containment. The reanalysis using CESEC-III assumed critical flow through the break consistent with the original analysis but accounted for the letdown line losses.

During the course of the review, the staff requested the licensee to address its compliance with the applicable restrictions specified in the NRC safety evaluation report (SER) (Reference 3) for use of the CESEC-III code. In response, the licensee indicated that: (1) the letdown line break event methodology using CESEC-III was previously approved (Reference 3), (2) both the original (Reference 4) and revised analyses used the previously approved Henry-Fauske correlation for the critical flow calculation, and (3) the results of the letdown line analysis, showing that no voiding existed in the reactor upper head and no two-phase flow was involved in the RCS cold-leg, were within the applicable range of the CESEC-III code.

Since the licensee has confirmed that the thermal-hydraulic conditions of the letdown line break analysis are within the applicable range of the approved CESEC-III code, and adequately addresses its compliance with the applicable SER restrictions (Reference 3), the staff concludes that the licensee’s use of CESEC-III for the letdown line break analysis is acceptable.

3.2.2 Use of the Core Protection Calculator (CPC) Hot-Leg Saturation Reactor Trip for Consequence Mitigation

The reanalysis credited the CPC hot-leg saturation trip and low pressurizer pressure trip for event mitigation, as compared to the CPC low departure from nucleate boiling ratio (DNBR) trip and low pressurizer pressure trip credited in the original analysis (Reference 4). The CPC hot-leg saturation trip occurs when the hot-leg temperature and pressure indicate that the saturation conditions have been reached in the RCS hot-legs. The reactor trips credited in both Reference 4 and the reanalysis are safety-related and redundant.

The NRC regulatory requirements-related inclusion of a limiting condition for operation (LCO) in the TSs are set forth in 10 CFR 50.36(c)(2)(ii). Specifically, Criterion 3 of 10 CFR 50.36(c)(2)(ii) states that an LCO is required for “[a] structure, system, or component that is a part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.” During the review, the staff requested the licensee to address its compliance with Criterion 3 of 10 CFR 50.36(c)(2)(ii) for use of the CPC hot-leg saturation trip in the reanalysis. In response, the licensee indicated (in Response 3 of Reference 2) that the CPC hot-leg saturation margin is a function of hot-leg temperature and pressurizer pressure. TS LCO 3/4.2.8 requires that the pressurizer pressure be between 2,050 pounds per square inch, absolute (psia) and 2,275 psia. The hot-leg temperature is dependent on the cold-leg temperature and the RCS flow rate. TS LCO 3/4.2.5 requires that the total RCS flow rate be greater than or equal to 148×10^6 pounds mass per hour (lbm/hr) and TS LCO 3/4.2.6 requires that the cold-leg temperature be between 541 ^{o}F and 558 ^{o}F. Maintaining these LCOs inherently preserves the CPC hot-leg saturation margin. The maximum cold-leg temperature corresponds to a saturation pressure of 1115.4 psia and the minimum pressurizer pressure corresponds to a saturation temperature of 637.6 ^{o}F. The corresponding saturation temperatures and pressures are outside the LCO ranges and require actions if exceeded.

Also, as discussed in the Bases for TS 2.1.1, the CPC quality margin - low (i.e., CPC hot-leg saturation) limit and the other limits listed are integral parts of ensuring a valid DNBR trip. Operation within the specified limits is constantly verified by the CPCs and a CPC auxiliary trip is generated if a limit is exceeded. TS LCO 3.3.1 (item 14 in TS Table 3.3-1) requires the CPCs to be operable. Therefore, TS LCO 3.3.1 ensures that the CPC auxiliary trip (i.e., CPC hot-leg saturation trip) credited in the revised letdown line break analysis is operable by ensuring that the CPCs are operable.

Based on the licensee's response and the TS LCO 3.3.1 requirements for CPCs operability, the staff has determined that the existing TS LCOs are adequate to ensure the CPC hot-leg saturation margin to be maintained, and, therefore, concludes that the licensee has satisfactorily addressed the staff concern regarding the compliance with the 10 CFR 50.36(c)(2)(ii) requirements.

3.2.3 RCS Mass Release Calculation and the Assumed Initial Conditions

As indicated in Section 3.1 of Reference 1, the licensee considered the following plant conditions for the letdown line break analysis:

- a maximum core power of 3480.6 megawatts thermal (MWt) (increased from 3478 MWt in the current analysis)
- a full-power core inlet coolant temperature of 560 ^{o}F (increased from 557.5 ^{o}F)
- a full-power RCS pressure of 2300 psia (increased from 2250 psia) and
- a full-power RCS flow rate of 170.2X10^{6} lbm/hr (increased from 148X10^{6} lbm/hr)

The values for the initial power level, reactor core inlet coolant temperature, and RCS pressure and flow used in the reanalysis were increased to the maximum allowable values for the plant safe operation. The difference in power level is due to a slight increase in the assumed reactor coolant pump heat load into the RCS. The licensee indicated that the initial plant conditions used in the analysis were determined to maximize the total RCS mass release. While the increase in the values for initial power level, reactor core inlet coolant temperature, and RCS pressure will increase the letdown line break flow rate and result in an increase in the RCS mass release, the increased RCS flow will increase the heat removal capability from the RCS primary-to-secondary side and result in a decrease in the RCS pressure; thus, a smaller RCS release. During the course of the review, the staff requested the licensee to justify adequacy of the use of a higher initial RCS flow in calculating the RCS mass release. In response, the licensee indicated (Reference 2) that the higher RCS flow would have a lower mass release if the consequences of the letdown line break analysis were independent of the time of reactor trip. In the reanalysis, the CPC hot-leg saturation trip was credited for reactor trip to terminate the event. The CPC hot-leg saturation trip is dependent upon hot-leg temperature. A higher

RCS flow results in a lower hot-leg temperature and delays the reactor trip. The licensee performed a sensitivity analysis between the minimum and maximum RCS flow and confirmed that the maximum RCS flow produced a higher break mass release prior to the CPC hot-leg saturation trip. Based on the results of the licensee's sensitivity analysis, the staff concludes that the assumed initial conditions for the letdown line break analysis will result in a maximized RCS mass release and are acceptable.

In the reanalysis, a letdown line isolation valve closure time of 10 seconds (increased from 5 seconds) was used. This change is to reflect the TRM specified valve closure time and is acceptable.

The reanalysis credited the reactor trip on the CPC hot-leg saturation trip signal and the valve isolation on the safety injection actuation signal to terminate the break flow. Using the plant initial conditions and the reactor trip signal discussed above, the licensee considered various break sizes and a delay time of up to 30 minutes for the automatic reactor trip and valve isolation in determination of the limiting case. The 30-minute delay time is consistent with the typical operator action time and is a reasonable maximum delay time assumed in the analysis for the automatic trip and isolation because operator action can be credited to trip the reactor and isolate the break after 30 minutes following the event initiation.

The results of the licensee's analysis indicated that the limiting case, resulting in the largest mass and radiological release, was a break size of 0.0094 ft^2 when the reactor trip and valve isolation occurred at 30 minutes after the event initiation.

The staff finds that the limiting break size was identified using adequate assumptions, and the RCS mass release calculation was based on the limiting break case, as discussed above, and therefore, concludes that the results of the mass release analysis are acceptable for the licensing applications.

3.2.4 Summary

Based on the review discussed in Section 3.2 above, the staff finds that: (1) the letdown line break reanalysis was performed with the previously NRC-approved computer code, (2) the licensee satisfactorily addressed its compliance with the applicable restrictions specified in Reference 3 for use of the approved computer code, and (3) the assumptions used in the reanalysis, resulting in a largest mass and radiological release, are adequate. Therefore, the staff concludes that the results of analysis for the letdown line break event are acceptable for use in calculating the radiological releases and to support the changes to Waterford 3 TS 3.4.7.

Table 1

Waterford 3 Letdown Line Break Accident Analysis Parameters

Core Power Level, MWt	3559.5
Reactor Coolant Equilibrium Activity, µCi/gm DEI-131	1
Reactor Coolant Maximum Activity, µCi/gm DEI-131	60
Reactor Coolant Radioisotope Concentrations From FSAR Table 11.1-2	
Mass of Coolant Released, lbm	67,000
Duration of Spill, minutes	30
Fraction of Iodines Immediately Released	0.4
Atmospheric Dispersion Factors, sec/m^3	
Exclusion Area Boundary (EAB), 0 -2 hr	6.3×10^{-4}
Low Population Zone (LPZ), 0 - 8 hr	7.1×10^{-5}

Table 2

Waterford 3 Letdown Line Break Accident, Licensee Dose Results

RCS Activity Scenario	EAB Thyroid Dose (rem)	LPZ Thyroid Dose (rem)	Thyroid Dose Acceptance Criteria (rem)	EAB Whole Body Dose (rem)	LPZ Whole Body Dose (rem)	Whole Body Dose Acceptance Criteria (rem)
TS Equilibrium	5	1	30	0.3	0.05	2.5
Induced Spike	30	8	30	0.4	0.1	2.5
Existing Spike	200	25	300	1	0.2	25

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 no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (67 FR 66009, dated October 29, 2002). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. W3F1-2001-0088, letter from J. Herron (Entergy) to NRC, "Technical Specification Change Request NPF-38-239, Revision of Letdown Line Break Dose Consequences," dated October 15, 2001

2. W3F1-2002-0072, letter from K. Peters (Entergy- Waterford 3) to NRC, "Supplement to Amendment Request NPF-38-239 Revision of Letdown Line Break Dose Consequences (TAC No. MB3231)," dated August 27, 2002.

3. NRC Safety Evaluation Report, "CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," April 3, 1984.

4. NUREG-0787, "Safety Evaluation Report related to the Operation of Waterford Steam Electric Station, Unit No. 3," July 1981.

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