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W3F1-2002-0072

August 27, 2002

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Waterford Steam Electric Station, Unit 3
Docket No. 50-382
Supplement to Amendment Request NPF-38-239 Revision of Letdown Line Break Dose Consequences (TAC NO. MB3231)

REFERENCES:
1. Entergy Letter dated October 15, 2001, "Technical Specification Change Request NPF-38-239 Revision of Letdown Line Break Dose Consequences" (W3F1-2001-0088)
2. NRC Letter Dated June 20, 2002, "Waterford Steam Electric Station, Unit 3- Request For Additional Information Related To Technical Specification Change Regarding Revision Of Letdown Line Break Dose Consequences (TAC No. MB3231)"

Dear Sir or Madam:

By letter (reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Waterford Steam Electric Station, Unit 3 (Waterford 3) Technical Specifications and Final Safety Analysis Report (FSAR) to utilize a new letdown line break dose analysis. In reference 1, Entergy 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 acceptance criterion of a small fraction (10%) of 10 CFR 100 limits was exceeded and NRC (Nuclear Regulatory Commission) approval was required in accordance with 10 CFR 50.59.

On May 10, 2002 Entergy and members of your staff participated in a conference call to discuss the number of charging pumps assumed to be in operation for the analysis. During this call, the NRC staff stated that an initial assumption of a single operating pump provides a suitable licensing basis analysis and has sufficient conservatism to accommodate two and three pump operating scenarios that may exist during the operating cycle. The NRC staff issued a request for additional information (reference 2) following the conference call

The results of the one pump analysis, provided in response to question seven of reference 2, fall within the acceptance criteria contained in the Standard ReviewPlan but the increase in dose exceeds 10% of the difference between the currently approved dose and the regulatory limit. Therefore, Entergy continues to seek prior NRC approval

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of this new analysis in accordance with 10 CFR 50.59. Additionally, Entergy continues to request approval for the associated Technical Specification change.

Attachment 1 contains the response to the request for additional information (reference 2) utilizing the results of the analysis assuming one charging pump is in operation. Since all methodologies and assumptions utilized in reference 1, except for the number of charging pumps in operation and the associated letdown flow rate, are identical, only the changes and analysis results are provided in response to question 7.

The Technical Specification change requested in reference 1 remains unchanged.

Attachment 2 provides a revised no significant hazards considerations and environmental impact evaluation which replaces the no significant hazards considerations and environmental impact evaluation provided in reference 1 in their entirety.

There are no new commitments contained in this letter. A short extension beyond the August 19, 2002 response due date, specified in reference 2, was discussed and agreed to with the Project Manager for Waterford 3.

If you have any questions or require additional information, please contact D. Bryan Miller at 504-739-6692.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 27, 2002.

Sincerely,

K. J. Peters

KJP/DBM/cbh

Attachments:
1. Supplement to Amendment Request NPF-38-239 Revision of Letdown Line Break Dose Consequences
2. Revised No Significant Hazards Considerations and Environmental Impact Evaluation

cc: E.W. Merschoff, NRC Region IV
N. Kalyanam, NRC-NRR
J. Smith
N.S. Reynolds
NRC Resident Inspectors Office
Louisiana DEQ/Surveillance Division
American Nuclear Insurers

Attachment 1

To

W3F1-2002-0072

Supplement to Amendment Request NPF-38-239
Revision of Letdown Line Break Dose Consequences

Supplement to Amendment Request NPF-38-239
Revision of Letdown Line Break Dose Consequences

Question 1:

The reanalysis was performed with the CESEC-III code, while the code for the existing analysis was CEFLASH-4AS. Please provide a discussion to address the compliance with the applicable restrictions specified in the Nuclear Regulatory Commission (NRC) safety evaluation report for use of the CESEC-III code and verify that the thermal-hydraulic conditions of the analysis were within the applicable range of the approved code.

Response 1:

The NRC Safety Evaluation Report (SER) [reference 1] Section III, IV.6(d), and V found the letdown line break event model acceptable. The SER CESEC-III code restrictions and resolutions are as follows:

a. CESEC-III is not applicable for an anticipated transient without scram (ATWS) event [reference 1 Section I]. A more detailed Steam Generator (SG) model would be required.

The letdown line break is not an ATWS type event.

b. Additional approval for feedwater and main steam line breaks methodologies is required (CESSAR Appendix 15B and 15C approval) [reference 1 Section II].

The feedwater and main steam line break methodologies have been approved [reference 4]. The letdown line break event methodology was accepted [reference 1 Section III, IV.6(d), and V].

c. The thermal-hydraulics is limited to transients which do not result in two-phase fluid conditions in the cold legs of the reactor coolant system [reference 1 Section IV.1(b)].

The letdown line break event does not have two-phase cold leg flow.

d. The critical flow model must be Identified and justified [reference 1 Section IV.5].

The revised letdown line break transmittal [reference 3] Section 3.1 stated that only the analysis parameters that changed will be described. The original [reference 4] and revised analyses [reference 3] both used the Henry-Fauske correlation for the critical flow. Reference 1 Section IV.5 stated that the Henry-Fauske option was previously approved for use in CEFLASH-4AS and is acceptable for CESEC-III.

e. The steam generator assumptions must be identified and justified [reference 1 Section IV.6(e)].

The letdown line break event methodology was accepted [reference 1 Section III, IV.6(d), and V] The steam generator assumptions and inputs have a minimal affect on the accident consequences; the break flow and duration dominate the results.

f. The upper head modeling assumptions must be justified for analyses in which upper head voiding occurs [reference 1 Section IV.6(h)].

Upper head voiding does not occur for the letdown line break event.

The CESEC-III SER restrictions are met and the thermal hydraulic analysis conditions are within the applicable range of the approved code.

References:
1. NRC Safety Evaluation Report, "CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," April 3, 1984.
2. CENPD-107 through Supplement 6, "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System."
3. W3F1-2001-0088, "Technical Specification Change Request NPF-38-239, Revision of Letdown Line Break Dose Consequences," October 15, 2001.
4. NUREG-0787, "Safety Evaluation Report related to the operation of Waterford Steam Electric Station, Unit No. 3," July 1981.

Question 2:

The values for the initial power level, reactor coolant system (RCS) inlet temperature, pressure and flow used in the reanalysis were increased to the maximum values in the allowable range shown in Final Safety Analysis Report (FSAR) Table 15.0-4. The initial conditions were determined to maximize the total RCS mass release. While an increase in the values for the initial power level, RCS inlet temperature and pressure will increase the RCS flow and result in a increase in the RCS mass release. However, the increase in RCS flow will also increase the heat removal capability from the RCS primary to secondary side and result in a decrease in the RCS pressure and thus, a smaller RCS mass release. Please justify if the use of a higher initial RCS flow is a conservative assumption in the calculations to maximize the RCS mass release.

Response 2:

The higher RCS flow would have a lower mass release if the consequences were independent of the time of reactor trip. The core protection calculator (CPC) hot leg saturation trip terminates the event and the trip is dependent upon hot leg temperature. Hot leg temperature is minimized by higher RCS flow ($Q = m\ C_p \Delta T$), thus delaying the trip. A parametric analysis between minimum and maximum RCS flow was performed and determined that maximum flow produced a higher break mass release prior to the CPC hot leg saturation trip.

Question 3:

The reanalysis credited the Core Protection Calculator (CPC) hot leg saturation trip against the CPC low departure from nucleate boiling ration (DNBR) trip for the FSAR analysis. The NRC's regulatory requirements related inclusion of a Limiting Condition for Operation (LCO) in Technical Specifications (TS) 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." Please address how the reanalysis complies with Criterion 3 of 10 CFR 50.36(c)(2)(ii) for the CPC hot leg saturation trip.

Response 3:

The LCO for TS 3/4.2.4 pertains to DNBR margin. This LCO requires that the DNBR margin be maintained. The DNBR safety limit is described in TS 2.1.1.1 and TS Bases 2.2.1. TS Bases 2.2.1 states that the CPC DNBR algorithm will initiate a trip on low quality margin. The CPC quality margin – low is the CPC hot leg saturation margin trip credited in the letdown line break analysis.

In addition to the DNBR LCO, the CPC hot leg saturation margin is a function of hot leg temperature and pressurizer pressure. TS 3/4.2.8 LCO pertains to pressurizer pressure. This LCO requires that the pressurizer pressure be between 2025 psia and 2275 psia. The hot leg temperature is dependent upon the cold leg temperature and the RCS flow rate. TS 3/4.2.5 LCO pertains to RCS flow rate. This LCO requires that the total RCS flow rate be greater than or equal to 148×10^6 lbm/hr. TS 3/4.2.6 LCO pertains to RCS cold leg temperature. The LCO requires that the cold leg temperature be between 541 °F and 558 °F. Maintaining these LCOs inherently preserves the CPC hot leg saturation margin. The maximum cold leg temperature corresponds to a saturation pressure of 1115.4psia and the minimum pressurizer pressure corresponds to a saturation temperature of 637.6 °F. The corresponding saturation temperatures and pressures are outside the LCO ranges and would require action if exceeded.

Question 4:

The letdown charging flow of 144 gallons per minute (gpm), from three charging pumps, was assumed in the reanalysis. The charging flow was increased from 44 gpm for the FSAR analysis in order to maximize the total RCS mass release. However, a lower charging flow rate maximizes the fluid temperature at the break thereby resulting in a higher flashing fraction for the fluid at the break. This in turn maximizes the offsite dose release due to the increased steam release at the break. The licensee is requested to provide a discussion to address the effects of the flashing fraction and total mass release on the offsite dose release and show that the assumed higher letdown charging flow (of 144 gpm) results in a higher offsite dose release and thus, is conservative.

Response 4:

Note: Based on May 10, 2002 discussion with the NRC staff and Question 7 below, the letdown line break analysis has been revised to utilize a letdown charging flow of 44 gpm.

The maximum letdown/charging assumption has the potential to affect the transient primarily in three ways: break enthalpy, fuel activity release rate, and trip initiation.

The break enthalpy is taken to correspond to 560 °F for the duration of the event [reference 1, Attachment 3, Section 15.6.3.1.5.1.5]. This temperature does not take credit for cooling provided by the regenerative heat exchanger or maximum charging flow. Thus, the flashing fraction is maximized for the event duration.

The steady state fuel activity release rate is a function of the amount of system cleanup that occurs due to letdown flow and activity decay [reference 1, Attachment 1, Section 3.2]. The larger letdown flow and activity decay corresponds to a larger fuel activity release rate. For the accident induced iodine spiking doses, the fuel activity release rate is multiplied by a factor of 500 [reference 2, Section 15.6.2]. The maximum letdown flow maximizes the fission products released to the RCS, which in turn maximizes the offsite doses.

The trip initiation occurs when saturation of the coolant is reached (CPC hot leg saturation trip). The maximum charging delays the depressurization and correspondingly delays the trip [reference 1, Attachment 1, Section 3.1]. The delayed trip increases the time prior to letdown isolation and maximizes the break release.

References:
1. W3F1-2001-0088, "Technical Specification Change Request NPF-38-239, Revision of Letdown Line Break Dose Consequences," October 15, 2001.
2. NUREG-0800 Rev. 2, "Standard Review Plan," July 1981.

Question 5:

Provide a discussion of the results of DNBR calculations and demonstrate that the applicable acceptance fuel failure criteria in Standard Review Plan 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," are met.

Response 5:

Standard Review Plan (SRP) Section 15.6.2 [reference 1] Section II identifies the acceptance criteria as meeting 10 CFR 100 [reference 2] requirements as it relates to the radiological consequences. No DNBR specific acceptance criteria are listed in SRP Section 15.6.2 and the previously approved letdown line break event [reference 3] did not present DNBR results. The revised letdown line break event did not explicitly calculate the transient DNBR.

The limiting letdown line break occurs with offsite power available. With offsite power available all reactor coolant pumps (RCPs) are operating, temperature remains essentially constant, and pressure decreases slowly until the trip setpoint is reached. The approach to DNBR is slow so that for these conditions, the CPC DNBR trip would terminate the event prior to exceeding the specified acceptable fuel design limit (SAFDL). No DNBR violation (fuel failure) is expected to occur for this event.

References:
1. NUREG-0800 Rev. 2, "Standard Review Plan," July 1981.
2. 10CFR100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."
3. NUREG-0787, "Safety Evaluation Report related to the operation of Waterford Steam Electric Station, Unit No. 3," July 1981.

Question 6:

The proposed TS changes revise the LCO limits on specific activity of the reactor coolant. The licensee stated that the changes were based on the results of the letdown line break (LDLB) reanalysis. The licensee is requested to confirm that the LDLB event is the limiting event for establishing the acceptance limits for RCS specific activity. Considering that different methods and computer codes, values of input parameters were used in the LDLB reanalysis, the staff requests the licensee to provide information discussing all the events that were considered in determination of the limiting case, and discuss applicable analytical results to demonstrate that the LDLB reanalysis is limiting and conservative.

Response 6:

The license amendment request [reference 1] is proposing that the Waterford Steam Electric Station, Unit 3 (Waterford 3) Operating License be amended to conservatively limit RCS activity permitted by TS Action Statement 3.4.7.a to 60 μCi/gm at all power levels.

The specific activity action limit is used in the steam generator tube rupture (SGTR) and main steam line break (MSLB) analyses. The SRP [reference 2] does not require the pre-existing iodine spike analysis for the LDLB event and Entergy Operations, Inc. (Entergy) did not present this analysis on the original Waterford 3 docket [reference 3]. The Waterford 3 LDLB pre-existing iodine spike analysis was performed to validate the NRC SER results [reference 3] as part of the Waterford 3 corrective action process. During the LDLB dose validation, it was realized that the MSLB, SGTR, and LDLB events all assume the pre-existing iodine spike limit as 60 μCi/gm at all power levels. Thus, the license amendment request is intended to maintain the specific activity action limit consistent with the bounding analyses for these events. This request is a conservative change and does not reduce the margin of any affected analyses.

Reference:
1. W3F1-2001-0088, "Technical Specification Change Request NPF-38-239, Revision of Letdown Line Break Dose Consequences," October 15, 2001.
2. NUREG-0800 Rev. 2, "Standard Review Plan," July 1981.

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

Question 7:

On page 2 of 12 of the submittal, it states that the pre-accident letdown flow assumed in the development of the iodine spiking model depicted in FSAR Figure 15.1-75 was determined by you to be non-conservative. The reasoning was that the original model assumed the normal configuration of 1 charging pump being in operation, whereas during periods of elevated activity levels in the RCS, the letdown flow will be maximized for RCS cleanup in accordance with site off normal procedures. Therefore, two, or possibly three, charging pumps may be in operation. You revised the iodine spiking model to bound the letdown flow expected from 3 charging pumps being in operation.

The standard model of the accident-induced iodine spike (as documented in Standard Review Plan (SRP) 15.6.2 for small line breaks) assumes that the iodine appearance rate from the fuel rods to the primary coolant increases to a value 500 times greater than the appearance rate corresponding to the iodine concentration at the equilibrium value stated in the TS. The letdown flow rate used in the calculation of the accident induced iodine spike should therefore be based on normal operating conditions, which is, for Waterford 3, one charging pump in operation. By performing this calculation in this manner, the dose from an accident-induced spike should be shown to be below the acceptance criteria of a small fraction of Part 100 for offsite dose. The justification given in the submittal for the dose being higher than the SRP 15.6.3 acceptance criteria is not acceptable to the staff.

Response 7:

Based on this question, Entergy has revised the analysis to reflect the assumption of only one charging pump in operation. The following provides the revised assumptions and analysis results assuming one charging pump is in operation. Except for charging and letdown flow, the assumptions and methodologies remain unchanged from those presented in reference 1.

Parameter	Original Analysis (FSAR)	New Analysis
Accident Induced Iodine Spike Filtration Flow, gpm	44 (1 pump)	44 (1 pump)

For the accident induced iodine spiking doses, larger letdown flow produces more adverse consequences. The reason for the more adverse consequences is that the event is established from steady state conditions with the fuel activity release rate equal to the amount of cleanup that occurs due to letdown flow and activity decay. Thus, the larger letdown flow equates to a larger fuel activity release rate. However, the new analysis assumes only one charging pump is in operation instead of three pumps in operation as originally submitted. During a May 10, 2002 conference call, the NRC staff stated it was acceptable to assume only one charging pump is in operation for this analysis therefore, 44 gpm is used to represent this plant configuration. Based on the May 10, 2002 conference call with the staff, it is Entergy's understanding that a single pump analysis provides a suitable licensing basis analysis and has sufficient conservatism to accommodate two and three pump operating scenarios that may exist during the operating cycle.

The consequences of the original analysis (current FSAR), the new analysis, and the NRC staff SER results are compared to the SRP acceptance criteria in the tables below. All SRP acceptance criteria are met.

Exclusion Area Boundary Dose, rem

Event Scenario	Original Analysis (FSAR)	New Analysis	SER NRC Staff Results	SRP Acceptance Criteria
Thyroid, no spike	49	5	5.3	30
Thyroid, induced spike	140	30	16	30
Thyroid, existing spike	**	200	265	**
WB*, no spike	0.24	0.3	**	2.5
WB*, induced spike	0.33	0.4	**	2.5
WB*, existing spike	**	1	**	**

* - WB refers to whole body dose; ** – not reported

Low Population Zone Dose, rem

Event Scenario	FSAR	New Analysis	SER NRC Staff Results	SRP Acceptance Criteria
Thyroid, no spike	5.5	1	0.1	30
Thyroid, induced spike	15	8	0.25	30
Thyroid, existing spike	**	25	4.2	**
WB*, no spike	0.027	0.05	**	2.5
WB*, induced spike	0.036	0.1	**	2.5
WB*, existing spike	**	0.2	**	**

* - WB refers to whole body dose; ** – not reported

The new analysis results assuming only one charging pump in operation meet the SRP acceptance criteria of a small fraction of the 10CFR100 limits (30 rem).

Reference:

1. Entergy letter dated October 15, 2001, "Technical Specification Change Request NPF-38-239 Revision of Letdown Line Break Dose Consequences" (W3F1-2002-0088)

Question 8:

A control room habitability analysis is not done for this submittal. Considering that the revised Exclusion Area Boundary thyroid doses for the pre-existing iodine spike are calculated to be higher than that for the LOCA, which is the basis for the current control room habitability analysis, why weren't the control room doses calculated?

Response 8:

The loss of coolant accident (LOCA) off-site radiological consequences compared to the new letdown line break event demonstrates that the LOCA releases are much greater. The letdown line break event induced iodine spike consequences are used for this comparison. The letdown line break atmospheric dispersion factors would be the same or better than the LOCA dispersion factors because the letdown line break occurs in the reactor auxiliary building where the capability exists for it to be released through charcoal filtered ventilation systems. These charcoal filtered ventilation systems, the controlled area ventilation system and the reactor auxiliary building ventilation system, are not credited in the analysis. Thus, since the LOCA activity release and atmospheric dispersion factors bound the letdown line break event, the corresponding LOCA control room doses would also bound the letdown line break.

Event Scenario	LOCA#	Letdown Line Break
Thyroid, EAB	94.10	30
WB*, EAB	9.04	0.4
Thyroid, LPZ	39.31	8
WB*, LPZ	2.21	0.1
Control Room, Thyroid	13.77	**
Control Room, WB*	0.87	**
Control Room, Skin	21.36	**

* - WB refers to whole body dose, ** – not reported, # - from FSAR Table 15 6-18

Attachment 2

To

W3F1-2002-0072

Revised No Significant Hazards Considerations and Environmental Impact Evaluation

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Entergy Operations, Inc. (Entergy) is proposing that the Waterford Steam Electric Station, Unit 3 (Waterford 3) Operating License be amended to conservatively limit Reactor Coolant System (RCS) activity permitted by Technical Specification (TS) Action Statement 3.4.7.a to 60 μCi/gm at all power levels. Entergy also requests the approval of the revised Final Safety Analysis Report (FSAR) Section 15.6.3.1 letdown line break analysis.

An evaluation of the proposed change has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the standards in 10CFR50.92(c). A discussion of these standards as they relate to this amendment request follows:

1. Will the operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response:

   The proposed change to the Technical Specifications (TS) conservatively limits Reactor Coolant System (RCS) activity permitted by Action Statement 3.4.7.a to 60 μCi/gm at all reactor power levels. The proposed change to the Final Safety Analysis Report (FSAR) Section 15.6.3.1 revises the letdown line break accident analyses.

   The probability of a previously evaluated accident is not affected by this change because the pre-existing iodine spike is not an accident initiator and the new letdown line break accident analysis does not affect any plant Structure, Systems, or Component (SSC) but merely determines the consequences of the previously evaluated accident.

   The TS change is conservative in that it will reduce the accident consequences for events occurring at lower power levels. The new letdown line break accident analysis meets the original Safety Evaluation Report (SER) and current Standard Review Plan (SRP) acceptance criteria of a small fraction of the 10CFR100 limits.

   Therefore, this change does <u>not</u> involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Will the operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response:

   The probability of a new or different accident is not affected by this change because the new letdown line break accident analysis does not affect any plant Structure, Systems, or Component but merely determines the consequences of the previously evaluated accident.

Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Will the operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response:

The TS change is more limiting in that it will reduce the accident consequences for events occurring at lower power levels.

The new letdown line break accident analysis, assuming one operating charging pump, meets the original SER and current SRP acceptance criteria of a small fraction of the 10CFR100 limits. This single pump analysis provides a suitable licensing basis analysis and has sufficient conservatism to accommodate two and three pump operating scenarios that may exist during the operating cycle.

Therefore, based on the reasoning presented above, Entergy Operations has determined that the requested change does not involve a significant hazards consideration.

ENVIRONMENTAL IMPACT EVALUATION

An evaluation of the proposed amendment has been performed pursuant to 10CFR51.22(b), which determined that the criteria for categorical exclusion set forth in 10CFR 51.22 (c) (9) of the regulations are met. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described previously in the evaluation.

2. As discussed in the significant hazards evaluation, this change does not result in a significant change or significant increase in the radiological doses for any Design Basis Accident. The proposed license amendment does not result in a significant change in the types or a significant increase in the amounts of any effluents that may be released off-site.

3. The proposed license amendment does not result in a significant increase to the individual or cumulative occupational radiation exposure because this Technical Specification change is conservative in that it will reduce the accident consequences for events occurring at lower power levels. The new letdown line break accident analysis does not affect any plant Structure, Systems, or Component (SSC) but merely determines the consequences of the previously evaluated accident. The best estimate dose consequences remain bounded by the current Standard Review Plan acceptance criteria.