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W3F1-2004-0071

August 19, 2004

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

SUBJECT: License Amendment Request NPF-38-256
Supplement to Alternate Source Term Submittal
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES:
1. Entergy Letter W3F1-2003-0053, "License Amendment Request NPF-38-256, Alternate Source Term," July 15, 2004
2. Entergy Letter W3F1-2003-0074, "License Amendment Request NPF-38-249, Extended Power Uprate," November 13, 2003

Dear Sir or Madam:

Entergy Operations, Inc. (Entergy) requested approval of an amendment for Waterford Steam Electric Station, Unit 3 (Waterford 3) to revise its licensing basis source term in Reference 1. Entergy proposed to implement an Alternate Source Term (AST) as permitted by 10 CFR 50.67 for calculating accident offsite doses and doses to control room personnel. That submittal provided the dose consequence analysis results for seven events. The supplemental report provided in Attachment 1 of this letter includes the dose consequence analysis results for the four additional events:

- Reactor Coolant Pump Seized Rotor / Sheared Shaft
- Inadvertent Atmospheric Dump Valve Opening
- Excess Main Steam Flow with Loss of Offsite Power
- Letdown Line Break

Entergy will submit the results of its ongoing analyses for control room shine due to Large Break Loss of Coolant Accident (LOCA) by August 31, 2004. That is the one analysis remaining to complete the information required to support the Waterford 3 request to adopt AST.

The proposed change to implement the alternative source term was evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it was determined that the change involved no significant hazards considerations. The bases for this determination are not affected by the attached evaluations of the four additional events.

A001

An Extended Power Uprate Report (PUR) supporting an amendment request to increase the Waterford 3 rated thermal power from 3441 Mwt to 3716 Mwt was submitted in Reference 2. Consistent with the existing Waterford 3 licensing basis, the PUR presents doses to control room personnel for only the LBLOCA and Fuel Handling Accident (FHA). Following the submittal of Reference 2, Waterford 3 performed a tracer gas test to quantify the control room in-leakage. Control room habitability and compliance with 10 CFR 50, Appendix A General Design Criterion (GDC) 19 under the extended power uprate conditions are demonstrated by the dose consequence evaluations provided in the License Amendment Request proposing the AST. The PUR continues to provide an acceptable basis for the off-site dose consequences for the extended power uprate.

Entergy had previously requested approval of this amendment request by April 1, 2005. We now recognize that approval of portions of this amendment request is necessary to support approval of the extended power uprate amendment request as discussed above. Entergy therefore requests that those portions of this amendment request necessary to support the extended power uprate amendment request be approved in conjunction with the extended power uprate request which is targeted for NRC approval by January 31, 2005.

This submittal includes a new commitment as summarized in Attachment 2. If you have any questions or require additional information, please contact Jerry Burford at 601-368-5755.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 19, 2004.

Sincerely,

KJP/fgb

Attachments:
1. Supplemental Licensing Report for the Radiological Consequences of Accidents for the Waterford Steam Electric Station, Unit 3 Using Alternative Source Term Methodology
2. List of Regulatory Commitments

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Attachment 1

W3F1-2004-0071

Supplemental Licensing Report for the Radiological Consequences of Accidents for the Waterford Steam Electric Station, Unit 3 Using Alternative Source Term Methodology

Licensing Report for the Radiological Consequences of Accidents for the Waterford Steam Electric Station, Unit 3 Using Alternative Source Term Methodology

August 17, 2004

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1.0 RADIOLOGICAL CONSEQUENCES UTILIZING NUREG-1465 SOURCE TERMS

1.1. Introduction

Reference 1 submitted a license amendment request to implement an Alternate Source Term (AST) as permitted by 10CFR50.67 for calculating accident offsite doses and doses to control room personnel for Waterford 3. That submittal provided dose consequence analyses for events expected to be limiting, and noted that a second AST submittal would be made to provide the results of additional analyses. Accordingly, this submittal provides additional AST analyses and evaluations as described below.

Dose consequence analyses are provided for the following events:

- Reactor Coolant Pump (RCP) Seized Rotor/Sheared Shaft,
- Inadvertent Atmospheric Dump Valve Opening (IADVO),
- Excess Main Steam Flow with Loss of Off-Site Power (Excess Load), and
- Letdown Line Break.

The current Waterford 3 licensing basis for the radiological consequences analyses for accidents discussed in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR) is based on methodologies and assumptions that are primarily derived from Technical Information Document (TID)-14844 and other early guidance.

Regulatory Guide (RG) 1.183 provides guidance on application of AST in revising the accident source terms used in design basis radiological consequences analyses, as allowed by 10CFR50.67. Because of advances made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents, 10CFR50.67 allows holders of operating licenses to voluntarily revise the traditional accident source term used in the design basis accident radiological consequence analyses with ASTs.

The implementation of the AST methodology results in several major departures from the assumptions used in the existing Waterford 3 design basis analyses. First, the AST methodology categorizes the accident releases into eight groups while the TID-14844 assumptions address only three categories of radionuclides. Also, instead of an instantaneous release, as assumed in TID-14844, a phased release is assumed for the AST. Another change from the previous design basis is the assumption that radioiodine is predominantly cesium iodide (CsI), an aerosol that is more amenable to mitigation mechanisms, while TID-14844 assumed that the radioiodine was primarily elemental. Another significant change is the relatively large amount of cesium released from the core based on the AST methodology (30% of the core inventory) as compared to the TID-14844 release of 1% of the core inventory. This large amount of cesium becomes an important contributor to long-term dose due to the relatively long half-life of the ^{134}Cs and ^{137}Cs isotopes. These changes to the design basis assumptions have a potential impact on the following:

- Post Accident Sampling System,
- Post Accident Shielding,
- Accident Monitoring Equipment,
- Leakage Control,
- Equipment Qualification (EQ),
- Loss of Coolant Accident (LOCA) Control Room Shielding Dose, and

- Risk Impact.

Vital Area Access and Plant Shielding

NUREG-0737, Item II.B.2, required a design review of plant shielding, vital area access, and protection of safety related equipment to be used in post-accident operations. The original Waterford 3 evaluations performed to address this Three Mile Island action item were based on the TID-14844 source terms as documented in UFSAR Section 12.3A. Since the short term radiation levels resulting from implementing the AST methodology are lower than those resulting from the TID-14844 source terms, there is no impact on plant shielding, vital area access or equipment required to respond to the accident conditions. Therefore, these items have not been reevaluated using the AST methodology. In addition, since there are no plant modifications required by the implementation of the AST methodology, there are no new vital area access or equipment qualification requirements which require evaluation. The impact to the Post Accident Sampling System (PASS) has not been evaluated because the PASS has been removed from the Waterford 3 Technical Specifications. Due to the location of the Technical Support Center (TSC) within the Control Room envelope, habitability of the TSC is bounded by the control room habitability for postulated accident conditions. The TSC and the emergency operations facility continue to meet the requirements of IV.E.8 of Appendix E to 10CFR50 and the Three Mile Island Action Plan III.A.1.2 as given in NUREG-0737, Supplement 1

Equipment Qualification (EQ)

RG 1.183, Section 1.3.5, provides discussion on the use of AST and the environmental qualification of safety-related equipment. Appendix I of this regulatory guide provides further guidance on the assumptions for evaluating radiation doses for EQ.

Since the postulated increase in the post-accident integrated dose is long term, there are no adverse affects to equipment relied on to perform safety related functions immediately following an accident.

Section 1.3.5 of RG 1.183 states that the Nuclear Regulatory Commission (NRC) is assessing the effect of increased cesium releases on EQ doses to determine whether further action is warranted. Until this issue is resolved, the regulatory guide states that either the AST or TID-14844 assumptions may be used for performing equipment qualification analyses. The regulatory guide further states that no plant modifications are required to address the impact of the difference in AST versus TID-14844 source term characteristics on EQ doses pending the outcome of the NRC evaluation of the generic issue. Based on this guidance, Waterford 3 will maintain the current equipment qualification based on TID-14844.

Reference 2 provided the results of the initial screening of candidate Generic Issue 187. Review of this issue used the categories in Draft Management Directive 6.4, "Generic Issue Program," to categorize the issue as a compliance issue, a subset of another issue, a burden reduction issue, or a safety issue. The panel concluded that candidate generic issue should be dropped, as having no significant chance of meeting the incremental risk thresholds for back-fit.

Risk Impact

The proposed change does not modify the physical design or operation of the plant. The use of AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents and has no direct effect on the probability of the accident. The use of AST does

not result in any change to core damage frequency or to large early release frequency. There is no impact upon accident initiator frequencies or on equipment failure probabilities since there is no physical plant change associated with the adoption of AST as the licensing basis for Waterford 3 UFSAR radiological analyses. Since there are no relaxations involved in the implementation of the AST methodology at Waterford 3, there is no impact on severe accident management strategies.

Leakage Control

There is no impact on leakage control since no relaxations are involved in the implementation of the AST methodology at Waterford 3. The existing plant systems and controls will continue to perform their intended design function.

1.2. Common Analysis Inputs and Assumptions

Common analysis inputs and assumptions are described in Section 1.2 of Attachment 2 of Reference 1. Some inputs and assumptions are identified therein as being specific to events evaluated in Reference 1 and are not applicable to the analyses presented in this submittal.

1.3. Control Room Air Conditioning System and Control Room Ventilation Model

The control room air conditioning system and control room ventilation model are described in Section 1.3 of Attachment 2 of Reference 1. The description includes event-specific control room unfiltered in-leakage assumptions for the events evaluated in Reference 1 which are not applicable to events evaluated in this submittal. The in-leakage assumptions for events evaluated in this submittal are:

Sequence Type	Control Room Unfiltered In-leakage Modeled
RCP Seized Rotor/Sheared Shaft	150 CFM
IADVO	100 CFM
Excess Load	150 CFM
Letdown Line Break	150 CFM

1.4. Exceptions to Reg. Guide 1.183

Exceptions applicable to this submittal are identified in Section 1.4 of Attachment 2 of Reference 1.

2.0 CONCLUSIONS

A summary of the calculated dose consequences of each analyzed scenario is presented in Table 2-1. All accident radiological consequences meet the acceptance criteria for the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Main Control Room (MCR). IADVO and Letdown Line Break results are given based on iodine spiking scenario, Pre-existing Iodine Spike (PIS), or Event Generated Iodine Spike (GIS).

TABLE 2-1
SUMMARY OF RESULTS

Event Scenario	Dose Consequences			Acceptance Criterion
	EAB	LPZ	MCR	EAB&LPZ/MCR
RCP Seized Rotor/Sheared Shaft	0.552	0.213	3.19	2.5/5
IADVO - PIS	0.222	0.077	2.203	2.5/5
IADVO - GIS	0.226	0.124	3.616	2.5/5
Excess Load	0.33	0.12	1.87	2.5/5
Letdown Line Break - PIS	4.217	0.665	0.997	25/5
Letdown Line Break - GIS	0.633	0.098	0.138	2.5/5

Notes: All Results are presented in units of rem Total Effective Dose Equivalent (TEDE).

Detailed discussions for each individual event are presented in Sections 4 through 11. The detailed analyses for each event demonstrate that radiological consequences meet the TEDE dose acceptance limits for off-site dose. The radiological consequences for MCR dose for all events are ≤ 5 Rem TEDE.

3.0 REFERENCES

1. W3F1-2004-0053, "License Amendment Request NPF-38-256, Alternate Source Term, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38," July 15, 2004.

2. Letter, J Rosenthal (NRC) to A. Thadani (NRC), "Initial Screening of Candidate Generic Issue 187, 'The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump,'" April 30, 2001.

3. W3F1-2004-0068, "Supplement to Amendment Request NPF-38-249, Extended Power Uprate, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38," August 10, 2004.

4. W3F1-2003-0074, "License Amendment Request NPF-38-249, Extended Power Uprate, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38," November 13, 2003.

5. Letter, N. Kalyanam (NRC) to J. Venable (Entergy), "Waterford Steam Electric Station Unit 3 – Issuance of Amendment Re: Letdown Line Break Dose Consequences Revision (TAC No. MB3231)," January 8, 2003.

4.0 REACTOR COOLANT PUMP (RCP) SEIZED ROTOR/SHEARED SHAFT

For the RCP Seized Rotor/Sheared Shaft dose analysis per RG 1.183 (Appendix G), the only fission product release path needed to be considered is fission product releases via steam generator (SG) steaming (releases via ADVs or Main Steam Safety Valves (MSSVs)). Per RG 1.183 Table 6, this pathway is analyzed until cold shutdown is established.

The secondary steaming pathway assumes that a RCP shaft has "locked up" rendering the RCP inoperable and that releases are occurring due to primary-to-secondary leakage from the Reactor Coolant System (RCS) to the SG. Activity is then released to the environment through the use of the ADVs to remove decay heat and to cool the plant to shutdown cooling entry conditions. Since the control room χ/Q values are worst for ADV releases, any releases which would occur through the MSSVs are instead assumed released from the ADV locations. Once shutdown cooling is initiated, the release to the environment is terminated for this pathway.

For purposes of radiological analyses, the analysis model for RCP Seized Rotor/Sheared Shaft is very similar to that for the Control Element Assembly (CEA) Ejection event.

4.1. Input Parameters and Assumptions

The input parameters and assumptions for the RCP Seized Rotor/Sheared Shaft analysis are listed in Table 4-1. Certain inputs and assumptions are discussed in additional detail below.

4.1.1. Source Term

As a result of the RCP Seized Rotor/Sheared Shaft accident, a maximum 15% fuel failure is allowed, as discussed in Reference 3. The Extended Power Uprate (EPU) Licensing Amendment Request (Reference 4) had originally reported an 8% fuel failure limit. For the purpose of the RCP Seized Rotor/Sheared Shaft analysis, a conservative range of fuel failures was analyzed (up to 15%). The maximum assumed fuel failure mechanism is up to 15% of the fuel rods in the core experiencing Departure from Nucleate Boiling (DNB). The non-LOCA gap fractions specified in Table 3 of RG 1.183 (Footnote 11) are conservatively selected for use in the RCP Seized Rotor/Sheared Shaft analysis to provide the most conservative set of results. These gap fractions are 10% for iodines and noble gases and 12% for alkali metals. Thus, this analysis conservatively applies the larger gap fractions required for CEA Ejection rather than the smaller gap fractions in the table itself which would apply for an RCP Seized Rotor/Sheared Shaft event.

4.1.2. Iodine Chemical Form

For the secondary steaming release pathway, the iodine releases from the SG to the environment are assumed to be 97% elemental iodine and 3% organic iodine. This is consistent with the guidelines provided in Appendix G of RG 1.183.

4.1.3. Release Pathways

Conservatively, all the iodine, alkali metal and noble gas activity due to the postulated RCP Seized Rotor/Sheared Shaft accident is assumed to be in the primary coolant when determining the dose consequences due to primary-to-secondary SG leakage and subsequent secondary steaming. Releases are assumed to be terminated once shutdown cooling is initiated and the SGs are no longer providing decay heat removal (no further releases would occur due to cooldown to cold shutdown conditions). This is consistent with the guidelines provided in RG 1.183, Table 6.

For the purposes of the RCP Seized Rotor/Sheared Shaft analysis, a primary-to-secondary SG leakage of 150 gpd is assumed. This value is consistent with the Technical Specification change to which Entergy committed in Reference 1 and is conservative with respect to a planned Technical Specification change request to further reduce this limit to 75 gpd.

Operator actions to select the more favorable of the control room air intakes in terms of χ/Q are credited to occur after 2 hours.

4.1.4. Removal Coefficients

For the secondary steaming path, iodine and alkali metal releases to the secondary side via primary-to-secondary side SG leakage are assumed to be subject to a Partition Factor (PF). Consistent with RG 1.183, Section 5.5, a PF of 100 is assumed for iodines and alkali metals. For the sake of conservatism, a PF of 10 is assumed for the first 30 minutes of the event to account for potential elevated releases due to the initial transient.

Per RG 1.183, all noble gases released to the secondary side via primary-to-secondary side SG leakage are assumed to be immediately released to the environment.

4.1.5. Main Control Room Model

The MCR ventilation model is described in Section 1.3 of Reference 1. The RCP Seized Rotor/Sheared Shaft dose model for secondary steaming release pathway conservatively assumes a constant unfiltered in-leakage of 150 CFM for the entire duration of the secondary steaming release (7.5 hours). The RCP Seized Rotor/Sheared Shaft analysis assumes that the preferred control room intake is selected at two hours into the event.

4.2. Results

Acceptance limits for the RCP Seized Rotor/Sheared Shaft transient per RG 1.183 Table 6, are:

- EAB and LPZ: 2.5 Rem TEDE
- MCR: 5 Rem TEDE

The radiological consequences in terms of Rem TEDE are listed below as a function of 15% fuel failure for an assumed primary-to-secondary SG leakage of 150 gpd and a control room unfiltered in-leakage of 150 CFM.

15% Fuel Failure	Secondary Steaming Release Pathway	Acceptance Criteria
EAB (worst two hour dose)	0.552	2.5 Rem TEDE
LPZ (duration)	0.213	2.5 Rem TEDE
MCR	3.19	5 Rem TEDE

Thus, the radiological consequences for the RCP Seized Rotor/Sheared Shaft are < 2.5 Rem TEDE for the EAB and LPZ doses and < 5 Rem TEDE for the MCR, based on 15% fuel failure, a 150 gpd primary-to-secondary leak rate per SG and maximum control room unfiltered in-leakage of 150 CFM in recirculation mode.

TABLE 4-1
ASSUMPTIONS USED FOR RCP SEIZED ROTOR/SHEARED SHAFT RADIOLOGICAL ANALYSIS

Core Power Level:	3735 MWt
Core Inventory:	See Table 1-1 of Reference 1
Fission Product Gap Fractions:	
Iodines	10%
Noble Gases	10%
Alkali metals (Cs & Rb-86)	12%
Fraction of Fuel Rods in Core Failing (maximum):	15%
<u>Secondary Steaming Pathway</u>	
Primary-to-Secondary Leak Rate:	150 gpd per SG
Iodine Chemical Form (Reactor Building Release Path):	
Elemental	97%
Organic	3%
Particulate	0%
Steaming PF (Iodine and Alkali Metals):	
0-30 minutes	10
> 30 minutes	100
Duration of Release:	7.5 hours
Control Room Parameters	See Table 1-2 of Reference 1

TABLE 4-1 (Cont.)
ASSUMPTIONS USED FOR RCP SEIZED ROTOR/SHEARED SHAFT RADIOLOGICAL ANALYSIS

Main Control Room χ/Q Assumed:

Time	Secondary Steaming Unfiltered In-leakage	Secondary Steaming Pressurization Flow
0-2 hr	1.06E-01	NA
2-8 hr	7.45E-02	2.08E-04 *

* factor of 4 reduction credited per SRP 6.4.

Steaming (lbm) and Activity (DEI-131, Ci) Releases

0-2 hr Steaming	2-8 hr Steaming
609,744	858,838

0-15 min	15-30 min	½-1 hr	1-2 hr	2–4 hr	4–6 hr	6-7.5 hr
2.70	3.54	1.73	6.03	17.73	23.16	19.56

Alkali Metal Source Term Data, Ci Released:

Cs-134	18.506
Cs-136	4.866
Cs-137	9.855
Rb-86	0.035

5.0 INADVERTENT ATMOSPHERIC DUMP VALVE OPENING (IADVO)

For the IADVO event, a single ADV is assumed to open. The radiological analysis for this event is constructed to be applicable for both the IADVO, as described in Section 2.13.1.1.4 of the EPU Licensing Amendment Request (Reference 4), and the IADVO with Single Active Failure (Loss of Offsite Power (LOOP) assumed), as described in Section 2.13.1.2.4 of the EPU Licensing Amendment Request. In each case, the ADV of the unaffected SG is used to cooldown the plant after reactor trip until shutdown cooling is initiated.

The IADVO will be documented/quantified in terms of the dose assessment for the Feedwater Line Break (FWLB) event. That is, the releases are assumed to be bounded by the releases associated with the analysis presented in Section 9.0 of Reference 1. The IADVO sequences are quantified in terms of the FWLB since the plant response and accident progression characteristics for these two events are similar, as modeled for dose analyses. The faulted or affected SG share sufficient plant response characteristics that their releases can be quantified as a single release applicable to both scenarios. In both cases, the unaffected SG is used to cooldown and depressurize the primary system to cold shutdown conditions using the ADV to remove decay heat. However, for the FWLB, the unaffected SG undergoes significant depletion of its inventory prior to reactor trip. In the FWLB event, the SG time constant is calculated using the mass versus time information. Additionally, the iodine decontamination factor is held at unity for the four hours it takes for the "unaffected" SG to have its tubes covered. The operators are assumed to not commence the cooldown of the RCS until this 4 hour time period. The model maintains 4 pumps in operation until the plant reaches 500 °F in this delayed cooldown. Therefore, it is conservative to use the FWLB releases from the unaffected SG for this analysis. No fuel failure is predicted for the IADVO or the FWLB.

The analysis considers the secondary steaming pathway from the intact SG. This is a minor contributor compared to the releases from the affected SG, contributing less than 1% to the overall doses. The operators are assumed to not commence cooldown for this event until 4 hours into the event, when the level has recovered to above the top of the U-tubes; this is consistent with the assumptions on PF. At that point, a cooldown of 50 °F/hr is assumed such that shutdown cooling is entered and releases from the intact SG are secured at 8 hours into the event. Note that the cooldown analysis neglects any cooldown that occurs due to the stuck open ADV. Also note that for IADVO, level would be recovered in significantly less time that the 4 hours assumed herein. That level recovery time is based on FWLB analyses. The doses due to releases from the affected and unaffected (intact) SGs are added to obtain the total dose.

5.1. Input Parameters and Assumptions

The input parameters and assumptions for the IADVO analysis are listed in Table 5-1. Certain inputs and assumptions are discussed in additional detail below.

The results of this analysis are developed to be applicable for both Hot Full Power (HFP) and Hot Zero Power (HZP) conditions. A HZP steam generator inventory is assumed for radiological release calculations to maximize the release; however, assumptions for PF are based on assuming the smaller HFP mass inventory.

5.1.1. Source Term

No fuel damage is predicted for the IADVO event (as is also the case for FWLB) as documented in the EPU Licensing Amendment Request. Two cases of iodine spiking are considered:

- A PIS case where a reactor transient is postulated to have occurred prior to the event and has raised primary coolant iodine concentration to the maximum value (60 μCi/gm DEI-131) allowed per Technical Specifications.

- A GIS case, where the primary system transient associated with the event causes an iodine spike in the primary system. A spiking factor of 500 is assumed.

A maximum RCS DEI-131 activity of 1.0 μCi/gm is assumed. Initial reactor coolant isotopic activity distribution is illustrated in Table 1-4 of Reference 1. This distribution is based on predicted activity distributions for power uprate conditions and is applicable for events for which no fuel failure will occur. The activity of Table 1-4 of Reference 1 is adjusted to correspond to the Technical Specification activity limit of 100/Ē μCi/gm.

A maximum Technical Specification secondary activity of 0.1 μCi/gm DEI-131 is assumed.

Since no fuel failure is postulated, the small contribution to dose consequences from alkali metals has been ignored.

5.1.2. Iodine Chemical Form

Per RG 1.183, iodine releases from the SGs to the environment are assumed to be 97% elemental and 3% organic. Since the same filter efficiency is specified for all iodine forms, there is no impact on the results of this assumption.

5.1.3. Release Pathways

Consistent with RG 1.183, Table 6, guidance for the Main Steam Line Break (MSLB) and the modeling for the FWLB, the IADVO event is analyzed for a release duration until cold shutdown is established. This corresponds to terminating releases at the point of shutdown cooling initiation for the unaffected SG, and a period of 36 hours for the affected SG. As discussed in Section 2.6.4.4 of Reference 4, Waterford 3 can achieve cold shutdown under natural circulation conditions within 36 hours of shutdown.

The release pathway for the intact SG is due to secondary steaming with radiological releases resulting from primary-to-secondary leakage from the RCS into the SGs. Activity is assumed to be released to the environment through the use of the ADVs to remove decay heat and to cool the plant to cold shutdown. Once cold shutdown is reached, the release to the environment is terminated. The doses due to releases from the affected and unaffected (intact) SGs are added to obtain the total dose.

The release pathway for the affected SG is due to primary-to-secondary leakage from the RCS to the secondary side of the SGs. Due to the postulated open ADV, activity transferred from the secondary side is assumed to be released immediately to the environment. Activity is assumed released until cold shutdown is achieved.

The primary-to-secondary leak rate for the intact SG is assumed to be 150 gpd. The primary-to-secondary leak rate for the faulted SG is assumed to be 540 gpd.

5.1.4. Removal Coefficients

A PF of 1.0 was assumed for the duration of the event for the affected SG. For the intact SG, a PF of 1.0 was assumed for the first 4 hours to account for the inventory loss due to the initial

transient and the time to recover the SG U-tubes. A PF of 100 for the non-faulted SG is subsequently assumed.

5.1.5. Main Control Room Model

The MCR ventilation model is described in Section 1.3 of Reference 1. The FWLB/IADVO dose model for secondary steaming assumes an unfiltered in-leakage of 100 CFM for the event duration. It is conservatively assumed that the pressurized mode is initiated at the start of the event and that the preferred control room intake is selected at two hours into the event. The filtered intake in the pressurized mode is assumed to be 225 CFM. The out-leakage from the control room envelope was assumed to be equal to the unfiltered in-leakage rate (100 CFM) during the first two hours and equal to the unfiltered (100 CFM) plus the minimum pressurization flow (50 CFM) for the accident duration.

5.2. Results

Acceptance limits for the IADVO transient per RG 1.183, Table 6 are:

- EAB and LPZ: 2.5 Rem TEDE
- MCR: 5 Rem TEDE

The radiological consequences in terms of Rem TEDE are listed below for an assumed primary-to-secondary SG leakage for the intact SG of 150 gpd and a control room unfiltered in-leakage of 100 CFM.

	TEDE Dose	Acceptance Criteria
PIS case:		
EAB (worst two hour dose)	0.222	2.5 Rem TEDE
LPZ (duration)	0.077	2.5 Rem TEDE
MCR	2.203	5 Rem TEDE
GIS case:		
EAB (worst two hour dose)	0.226	2.5 Rem TEDE
LPZ (duration)	0.124	2.5 Rem TEDE
MCR	3.616	5 Rem TEDE

Thus, the radiological consequences for the IADVO are < 2.5 Rem TEDE for the EAB and LPZ doses and < 5 Rem TEDE for the MCR. These are based on a 150-gpd primary-to-secondary leak rate for the unaffected SG and maximum control room unfiltered in-leakage of 100 CFM.

TABLE 5-1
ASSUMPTIONS USED FOR THE INADVERTENT ADV OPENING RADIOLOGICAL ANALYSIS

Core Power Level:	3735 MWt
Core Inventory:	See Table 1-1 of Reference 1
<u>Secondary Steaming Pathway</u>	
Primary-to-Secondary Leak Rate:	150 gpd unaffected SG, 540 gpd faulted SG
Iodine Chemical Form (Reactor Building Release Path):	
Elemental	97%
Organic	3%
Steaming PF (Iodine and Alkali Metals):	
4 hours	1
> 4 hours	100
Duration of Release:	36 hours
Control Room Parameters	See Table 1-2 of Reference 1

TABLE 5-1 (Cont.)
ASSUMPTIONS USED FOR THE INADVERTENT ADV OPENING RADIOLOGICAL ANALYSIS

Affected SG is Assumed to be the East SG.. MCR χ/Q Assumed:

Time	Unfiltered In-leakage, East ADV to East MCR Air Intake	Pressurization Flow, East ADV to East MCR Air Intake	Pressurization Flow, East ADV to West MCR Intake	Pressurization Flow, West ADV to East MCR Air Intake	Pressurization Flow, West ADV to West MCR Air Intake
0-2 hr	1.06E-01	1.06E-01	NA	1.36E-03	NA
2-8 hr	7.45E-02	NA	2.08E-04 *	NA	1.41E-03 *
8-24 hr	3.30E-02	NA	1.00E-04 *	NA	6.42E-04 *
1-4 d	2.31E-02	NA	6.58E-05 *	NA	5.10E-04 *

* factor of 4 reduction credited per SRP 6.4.

Steaming (lbm) and Activity (DEI-131, Ci) Releases

0-2 hr Steaming	2-8 hr Steaming
588,365	1,333,286

150 gpd Leakage

	0-15 min	15-30 min	½-1 hr	1-2 hr	2–4 hr	4–6 hr	6-8 hr
PIS	16.00	6.72	1.14	2.08	4.36	2.66	2.67
GIS	15.72	6.19	0.40	1.04	4.09	4.27	5.93

540 gpd Leakage

	0-15 min	15-30 min	½-1 hr	1-2 hr	2–4 hr	4–6 hr	6-8 hr
PIS	16.80	8.34	3.78	7.39	15.55	9.47	9.53
GIS	15.80	6.43	1.14	3.72	14.57	15.21	21.15

6.0 EXCESS MAIN STEAM FLOW (EXCESS LOAD)

An increase in main steam flow may be caused by any one of the following incidents:

- An inadvertent increase in the opening of the turbine admission valves caused by operator error or turbine load limit malfunction,
- Failure in the Steam Bypass System, which could result in an opening of one of the turbine bypass valves, or
- An inadvertent opening of an ADV or steam generator safety valve.

For the Excess Load with a LOOP dose analysis, the only fission product release path that needs to be considered is fission product releases via SG steaming (releases via ADVs or MSSVs). Per RG 1.183 Table 6, this pathway is analyzed until cold shutdown is established.

The secondary steaming pathway assumes that an Excess Load resulted in a reactor scram and that releases are occurring due to primary-to-secondary leakage from the RCS to the SG. Activity is then released to the environment through the use of the ADVs to remove decay heat and to cool the plant to shutdown cooling entry conditions. Since the control room χ/Q values are worst for ADV releases, any releases which would occur through the MSSVs are instead assumed released from the ADV locations. Once shutdown cooling is initiated, the release to the environment is terminated for this pathway.

For the Excess Load event, the reactor is assumed to be at a power operating limit due to the excess steam demand transient for some period of time prior to the postulated LOOP. As described in Section 2.13.1.2.3.1 of the EPU Licensing Amendment Request (Reference 4), it is assumed that a transient initially occurs that degrades all the thermal margin preserved by COLSS and brings the hot channel to DNB Ratio (DNBR) Specified Acceptable Fuel Design Limit (SAFDL) conditions. Because of this pre-trip power excursion, it is possible that the reactor could be operating above 100% rated thermal power for some period prior to the reactor trip on LOOP. The RPS includes a trip on Linear Power Level - High of 108% of rated thermal power, per Technical Specification Table 2.2-1. Thus, to account for the impact of this relatively short duration potential power excursion at the start of the event, a 10% factor will be applied to the calculated offsite and main control room doses.

6.1. Input Parameters and Assumptions

The input parameters and assumptions for the Excess Load analysis are listed in Table 6-1. Certain inputs and assumptions are discussed in additional detail below.

6.1.1. Source Term

As a result of the Excess Load, an 8% fuel failure was reported in the EPU Licensing Amendment Request. Therefore, the maximum assumed fuel failure mechanism is up to 8% of the fuel rods in the core experiencing DNB. The non-LOCA gap fractions specified in Table 3 (Footnote 11) of RG 1.183 are conservatively selected for use in the Excess Load analysis to provide the most conservative set of results. These gap fractions are 10% for iodines and noble gases and 12% for alkali metals. Thus, this analysis conservatively applies the larger gap fractions required for CEA Ejection rather than the smaller gap fractions in the table itself which would apply for an Excess Load event.

6.1.2. Iodine Chemical Form

For the secondary steaming release pathway, the iodine releases from the SG to the environment are assumed to be 97% elemental iodine and 3% organic iodine. This is consistent with the guidelines provided in RG 1.183.

6.1.3. Release Pathways

Conservatively, all the iodine, alkali metal and noble gas activity due to the postulated Excess Load transient is assumed to be in the primary coolant when determining the dose consequences due to primary-to-secondary SG leakage and subsequent secondary steaming. Releases are assumed to be terminated once shutdown cooling is initiated and the SGs are no longer providing decay heat removal (no further releases would occur due to cooldown to cold shutdown conditions). This is consistent with the guidelines provided in Table 6 of RG 1.183.

For the purposes of the Excess Load analysis, a primary-to-secondary SG leakage of 150 gpd is assumed. This value is consistent with the Technical Specification change to which Entergy committed in Reference 1 and is conservative with respect to a planned Technical Specification change request to further reduce this limit to 75 gpd.

Operator actions to select the more favorable of the control room air intakes in terms of χ/Q are credited to occur after 2 hours.

6.1.4. Removal Coefficients

For the secondary steaming path, iodine and alkali metal releases to the secondary side via primary-to-secondary side SG leakage are assumed to be subject to a PF. Consistent with RG 1.183, Section 5.5, a PF of 100 is assumed for iodines and alkali metals. For the sake of conservatism, a PF of 10 is assumed for the first 30 minutes of the event to account for potential elevated releases due to the initial transient.

Per RG 1.183, all noble gases released to the secondary side via primary-to-secondary side SG leakage are assumed to be immediately released to the environment.

6.1.5. Main Control Room Model

The MCR ventilation model is described in Section 1.3 of Reference 1. The Excess Load dose model for secondary steaming release pathway conservatively assumes a constant unfiltered in-leakage of 150 CFM for the entire duration of the secondary steaming release (7.5 hours). The Excess Load analysis assumes that the preferred control room intake is selected at two hours into the event.

6.2. Results

Acceptance limits for the Excess Load transient per RG 1.183, Table 6 are:

- EAB and LPZ: 2.5 Rem TEDE
- MCR: 5 Rem TEDE

The radiological consequences in terms of Rem TEDE are listed below as a function of 8% fuel failure for an assumed primary-to-secondary SG leakage of 150 gpd and a control room unfiltered in-leakage of 150 CFM.

8% Fuel Failure	Secondary Steaming Release Pathway	Acceptance Criteria
EAB (worst two hour dose)	0.33	2.5 Rem TEDE
LPZ (duration)	0.12	2.5 Rem TEDE
MCR	1.87	5 Rem TEDE

Thus, the radiological consequences for the Excess Load are < 2.5 Rem TEDE for the EAB and LPZ doses and < 5 Rem TEDE for the MCR, based on 8% fuel failure, a 150 gpd primary-to-secondary leak rate per SG and maximum control room unfiltered in-leakage of 150 CFM in recirculation mode.

TABLE 6-1
ASSUMPTIONS USED FOR THE EXCESS LOAD RADIOLOGICAL ANALYSIS

Core Power Level:	3735 MWt
Core Inventory:	See Table 1-1 of Reference 1
Fission Product Gap Fractions:	
Iodines	10%
Noble Gases	10%
Alkali metals (Cs & Rb-86)	12%
Fraction of Fuel Rods in Core Failing (maximum):	8%
<u>Secondary Steaming Pathway</u>	
Primary-to-Secondary Leak Rate:	150 gpd per SG
Iodine Chemical Form (Reactor Building Release Path):	
Elemental	97%
Organic	3%
Particulate	0%
Steaming PF (Iodine and Alkali Metals):	
0-30 minutes	10
> 30 minutes	100
Duration of Release:	7.5 hours
Control Room Parameters	See Table 1-2 of Reference 1

TABLE 6-1 (Cont.)
ASSUMPTIONS USED FOR THE EXCESS LOAD RADIOLOGICAL ANALYSIS

Main Control Room χ/Q Assumed:

Time	Secondary Steaming Unfiltered In-leakage	Secondary Steaming Pressurization Flow
0-2 hr	1.06E-01	NA
2-8 hr	7.45E-02	2.08E-04 *

* factor of 4 reduction credited per SRP 6.4.

Steaming (lbm) and Activity (DEI-131, Ci) Releases

0-2 hr Steaming	2-8 hr Steaming
609,744	858,838

0-15 min*	15-30 min*	½-1 hr*	1-2 hr*	2–4 hr*	4–6 hr*	6-7.5 hr*
1.74	2.04	0.96	3.29	9.55	10.80	10.47

* Note: a 10% transient overpower factor is required for this sequence, however these activity release rates do not reflect the transient overpower. Therefore, in order to account for this, the TEDE doses were conservatively increased by 10%.

Alkali Metal Source Term Data, Ci Released:

Cs-134	9.657 Ci
Cs-136	2.539 Ci
Cs-137	5.143 Ci
Rb-86	0.018 Ci

7.0 LETDOWN LINE BREAK

The letdown line break outside of containment provides a release path for the reactor coolant into the Reactor Auxiliary Building (RAB). The location of the postulated break is in an area of the RAB that is served by the normal RAB ventilation system. As a result, all releases to the environment from the break would be through the plant stack. During the first 30 minutes following the initiation of the transient, less than 78,000 lbm of reactor coolant is released from the ruptured letdown line. At this time operator action occurs to isolate the letdown line and trip the reactor. Prior to operator action there are no secondary steam releases through the MSSVs or ADVs.

The secondary steaming pathway assumes that releases are occurring due to primary-to-secondary leakage from the RCS to the SG. Activity is then released to the environment through the use of the ADVs to remove decay heat and to cool the plant to shutdown cooling entry conditions. At 30 minutes, the operators are assumed to initiate a controlled cooldown of the RCS until shutdown cooling entry conditions (350 °F nominal) are reached. Since the control room χ/Q values are worst for ADV releases, any releases which would occur through the MSSVs are instead assumed released from the ADV locations. Once shutdown cooling is initiated, the release to the environment is terminated for this pathway. Note that the releases from the primary system over the first 30 minutes of this event dominate.

7.1. Input Parameters and Assumptions

A summary of input parameters and assumptions is provided in Table 7-1. Certain assumptions are discussed in additional detail below.

7.1.1. Source Term

No fuel damage is predicted for the Letdown Line Break event as documented in the EPU Licensing Amendment Request (Reference 4). Two cases of iodine spiking are considered:

- A PIS case where a reactor transient is postulated to have occurred prior to the event and has raised primary coolant iodine concentration to the maximum value (60 μCi/gm DEI-131) allowed per Technical Specifications.

- A GIS case, where the primary system transient associated with the event causes an iodine spike in the primary system. A spiking factor of 500 is assumed.

A maximum RCS DEI-131 activity of 1.0 μCi/gm is assumed. Initial reactor coolant isotopic activity distribution is illustrated in Table 1-4 of Reference 1. This distribution is based on predicted activity distributions for power uprate conditions and is applicable for events for which no fuel failure will occur. The activity of Table 1-4 of Reference 1 is adjusted to correspond to the Technical Specification activity limit of 100/$\bar{E}$ μCi/gm.

A maximum Technical Specification secondary activity of 0.1 μCi/gm DEI-131 is assumed.

Since no fuel failure is postulated, the small contribution to dose consequences from alkali metals has been ignored.

7.1.2. Iodine Chemical Form

Per RG 1.183, iodine releases from the SGs to the environment are assumed to be 97% elemental and 3% organic. Since the same filter efficiency is specified for all iodine forms, there is no impact on the results of this assumption.

7.1.3. Release Pathways

Conservatively, all the iodine and noble gas activity due to the postulated Letdown Line Break is assumed to be in the primary coolant when determining the dose consequences due to primary-to-secondary SG leakage and subsequent secondary steaming. Releases are assumed to be terminated once shutdown cooling is initiated and the SGs are no longer providing decay heat removal (no further releases would occur due to cooldown to cold shutdown conditions). This is consistent with the guidelines provided in Table 6 of RG 1.183. Based on plant scram occurring at 30 minutes into the event and allowing 7.5 hours for subsequent cooldown, the radiological analysis assumes that shutdown cooling is initiated at 8 hours into this event.

The activity released from the ruptured letdown line is assumed to be released directly to the environment through the plant stack during the time period immediately following the accident and before isolation (0-30 minutes). The location of the postulated break is in an area of the RAB that is served by the normal auxiliary building ventilation system. As a result, all releases to the environment from the break would be through the plant stack.

For the purposes of the Letdown Line Break analysis, a primary-to-secondary SG leakage rate of 150 gpd per SG is assumed. This value is consistent with the Technical Specification change to which Waterford 3 committed in Reference 1 and is conservative with respect to a planned Technical Specification change request to further reduce this limit to 75 gpd.

Operator actions to select the more favorable of the control room air intakes in terms of χ/Q are credited to occur after 2 hours.

7.1.4. Removal Coefficients

For the secondary steaming path, iodine releases to the secondary side via primary-to-secondary side SG leakage are assumed to be subject to a PF. Consistent with RG 1.183, Section 5.5, a PF of 100 is assumed for iodines. For the sake of conservatism, a PF of 10 is assumed for the first 30 minutes of the controlled cooldown (30 min – 1 hr) to account for potential elevated releases due to the initial transient.

Per RG 1.183, all noble gases released to the secondary side via primary-to-secondary side SG leakage are assumed to be immediately released to the environment.

7.1.5. Main Control Room Model

The MCR ventilation model is described in Section 1.3 of Reference 1. The Letdown Line Break dose model for secondary steaming assumes an unfiltered in-leakage of 150 CFM for the event duration. It is conservatively assumed that the pressurized mode is initiated at the start of the event and that the preferred control room intake is selected at two hours into the event. The filtered intake in the pressurized mode is assumed to be 225 CFM. The out-leakage from the control room envelope was assumed to be equal to the unfiltered in-leakage rate (150 CFM) during the first two hours and equal to the unfiltered (150 CFM) plus the minimum pressurization flow (50 CFM) for the accident duration.

7.2. Results

RG 1.183 Table 6 does not establish explicit radiological acceptance limits for the Letdown Line Break event. Acceptance limits for the Letdown Line Break established for Waterford 3 (Reference 5) are 10CFR100 limits for the PIS case and a small fraction (i.e., 10%) of 10CFR100 limits for this GIS case. This logic is assumed to be maintained for AST analyses, i.e., that the applicable limits are the 10CFR50.67 limits for the PIS case and a small fraction of the 10CFR50.67 limits for the GIS case. Thus, the limits are:

- EAB and LPZ (GIS case): 2.5 Rem TEDE
- EAB and LPZ (PIS case): 25 Rem TEDE
- MCR (PIS and GIS case): 5 Rem TEDE

The radiological consequences in terms of Rem TEDE are listed below for an assumed primary-to-secondary SG leakage of 150 gpd and a control room unfiltered in-leakage of 150 CFM.

	TEDE Dose	Acceptance Criteria
PIS case:		
EAB (worst two hour dose)	4.217	25 Rem TEDE
LPZ (duration)	0.645	25 Rem TEDE
MCR	0.997	5 Rem TEDE
GIS case:		
EAB (worst two hour dose)	0.633	2.5 Rem TEDE
LPZ (duration)	0.098	2.5 Rem TEDE
MCR	0.138	5 Rem TEDE

Thus, the radiological consequences for the Letdown Line Break are < 25 Rem TEDE for the EAB and LPZ doses and < 5 Rem TEDE for the MCR for the PIS case. Radiological consequences are < 2.5 Rem TEDE for EAB and LPZ doses and < 5 Rem TEDE for the GIS case. These are based on a 150 gpd primary-to-secondary leak rate for the unaffected SG and maximum control room unfiltered in-leakage of 150 CFM.

TABLE 7-1
ASSUMPTIONS USED FOR THE LETDOWN LINE BREAK RADIOLOGICAL ANALYSIS

Core Power Level:	3735 MWt
<u>Secondary Steaming Pathway</u>	
Primary-to-Secondary Leak Rate:	150 gpd for both SG
Iodine Chemical Form (Reactor Building Release Path):	
Elemental	97%
Organic	3%
Steaming PF (Iodine and Alkali Metals, Intact SG):	
0-30 minutes	10
> 30 minutes	100
Duration of Release:	8 hours
Control Room Parameters	See Table 1-2 of Reference 1

Main Control Room χ/Q Assumed:

Time	East MCR Air Intake	Unfiltered In-leakage, East ADV to East MCR Air Intake	Unfiltered In-leakage, West ADV to East MCR Air Intake	Pressurization Flow, East ADV to West MCR Intake	Pressurization Flow, West ADV to West MCR Air Intake
0-30 min	2.77E-03	NA	NA	NA	NA
30 min-2 hr	NA	1.06E-01	1.36E-03	NA	NA
2-8 hr	NA	7.45E-02	8.29E-04	2.08E-04 *	1.41E-03 *

* factor of 4 reduction credited per SRP 6.4.

TABLE 7-1 (Cont.)
ASSUMPTIONS USED FOR THE LETDOWN LINE BREAK RADIOLOGICAL ANALYSIS

Steaming (lbm) and Activity (DEI-131, Ci) Releases

0-2 hr Steaming	2-8 hr Steaming
794,217	1,357,617

	0-15 min	15-30 min	½-1 hr	1-2 hr	2–4 hr	4–6 hr	6-8 hr
PIS	404.81	360.15	0.70	0.12	0.23	0.21	0.24
GIS	23.00	48.33	0.06	0.11	0.19	0.20	0.27

Noble Gas Releases:

	0-15 min	15-30 min	½-1 hr	1-2 hr	2–4 hr	4–6 hr	6-8 hr
Kr-83m	806.68	717.98	2.1	4.57	10.1	10.28	10.28
Kr-85	16.45	14.65	0.05	0.1	0.21	0.21	0.21
Kr-85m	17.45	15.53	0.05	0.1	0.22	0.23	0.23
Kr-87	18.45	16.42	0.05	0.11	0.24	0.24	0.24
Kr-88	855.7	761.61	2.23	4.85	10.72	10.9	10.9
Xe-131m	73.86	65.74	0.2	0.42	0.93	0.95	0.95
Xe-133m	31.74	28.25	0.09	0.18	0.4	0.41	0.41
Xe-133	132.93	118.31	0.35	0.76	1.67	1.7	1.7
Xe-135m	69.04	61.45	0.18	0.4	0.87	0.88	0.88
Xe-135	62.48	55.61	0.17	0.36	0.79	0.8	0.8

Attachment 2

W3F1-2004-0071

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Entergy will submit the results of its ongoing analyses for control room shine due to Large Break Loss of Coolant Accident (LOCA) by August 31, 2004	X		August 31, 2004