Entergy

Entergy Nuclear Southwest
Entergy Operations, Inc.
17265 River Road
Killona, LA 70066
Tel 504 739 6660
Fax 504 739 6678

John T. Herron
Vice President, Operations
Waterford 3
jherron@entergy.com

W3F1-2001-0088
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PR

October 15, 2001

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Technical Specification Change Request NPF-38-239
Revision of Letdown Line Break Dose Consequences

Gentlemen:

Pursuant to 10CFR50.59 and 10CFR50.90, Entergy Operations, Inc. (Entergy) hereby requests an amendment of the Facility Operating License NPF-38 for Waterford Steam Electric Station, Unit 3 (Waterford 3). NRC staff review and approval is requested for the attached changes to the Technical Specifications (TS) and the Waterford 3 Final Safety Analysis Report (FSAR). New analyses of the letdown line break dose consequences were reviewed in accordance with 10CFR50.59. This review determined that the revised doses are within the regulatory limits of 10CFR100, but exceed the small fraction (10%) acceptance criterion given in the Standard Review Plan. On this basis, Entergy requests a license amendment in accordance with 10CFR50.90 requesting NRC staff approval of the revised letdown line break analyses. Marked up pages of the Waterford 3 TS and FSAR are attached to reflect the revised analyses. The attached description and safety analyses support the proposed changes to the Waterford 3 TS and FSAR.

The proposed change has been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in the attached submittal.

There are no commitments contained in this submittal.

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Entergy requests the approval of this change with an effective date within 60 days of approval. Although this request is neither exigent nor emergency, your prompt review is requested. Should you have any comments or questions please contact T. N. Schreckengast at (504) 739-6349.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 15, 2001.

Very truly yours,

J. T. Herron
Vice President, Operations
Waterford 3

JTH/WJS/TNS/cbh

Attachment 1: Description and No Significant Hazards Consideration Determination
Attachment 2: Proposed TS Marked-up
Attachment 3: Proposed FSAR Marked-up

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

W3F1-2001-0088

DESCRIPTION AND
NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

IN THE MATTER OF AMENDING

LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

1.0 DESCRIPTION OF PROPOSED CHANGES

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. This change will be implemented by deleting Figure 3.4-1 and incorporating the proposed limit into the action statement 3.4.7.a.

The proposed change to the Final Safety Analysis Report (FSAR) revises the letdown line break accident analyses. The re-analysis results were evaluated in accordance with 10CFR50.59 and the consequences were deemed to require prior NRC approval. The revised doses are within the regulatory limits of 10CFR100, but exceed the small fraction (10%) acceptance criteria given in the Standard Review Plan (SRP). The small fraction acceptance criteria was the basis used by the NRC staff for granting the license.

Also in this change, Entergy Operations, Inc. (Entergy) proposes to consolidate specification 3/4.4.7 such that the resulting specification is on consecutive pages and not interspersed with unnecessary blank pages. The blank pages will be moved to the end of the specification. These changes consist of:

1. Consolidating the information (i.e., Action 'c' and Surveillance Requirement 4.4.7) from page 3/4 4-25 onto the bottom of page 3/4 4-24.
2. Moving Table 4.4-4 to page 3/4 4-25.
3. Inserting a page that indicates that both pages 3/4 4-26 and 3/4 4-27 have been deleted. These pages currently contain Table 4.4-4, which is being moved, and Figure 3.4-1, which is being deleted.
4. The TS index pages are being revised to reflect the impact of the above changes.

1.1 Proposed Marked-up Specification

See Attachment 2 for the proposed TS change
See Attachment 3 for the proposed FSAR change

2.0 BACKGROUND

The Waterford Steam Electric Station Unit 3 (Waterford 3) Final Safety Analysis Report (FSAR) describes the analysis of the letdown line break event in Section 15.6.3.1. A condition report[1] was prepared to document a discrepancy between the letdown line break analysis assumed isolation time of 5 seconds and the Technical Requirements Manual (TRM) requirement of 10 second closure time for the isolation valve. During the re-analysis for the increased isolation valve stroke time, two additional issues arose and were documented in another condition report[2].

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[1] CR-WF3-1998-0274

[2] CR-WF3-1999-0591

The first issue was for the pre-existing iodine spiking (TS Figure 3.4-1) scenario in which a high RCS activity is assumed to exist at the initiation of the line break. The original letdown line break analysis did not perform a pre-existing iodine spike calculation. However, in the Waterford 3 Safety Evaluation Report (SER)[3], the NRC described their results for a 100% power case with a pre-existing iodine spike of 50 μCi/gm. The NRC calculation results demonstrated that the pre-existing iodine spike dose consequences remained within the 10CFR100 limits. During the Waterford 3 re-analysis, the pre-existing iodine spike was investigated for its effects on the letdown line break dose consequences. The historical industry approach has been to perform the FSAR Chapter 15 pre-existing spike calculations at 60 μCi/gm which corresponds to the maximum allowable pre-existing iodine spike at 100% power per TS Figure 3.4-1. However, TS Figure 3.4-1 allows the primary coolant activity to increase below 80% power such that the limiting dose consequences may not occur at 100% power, but at some lower power level due to the higher primary activity. The Waterford 3 letdown line break reanalysis performed a parametric on power and determined that the mass release was not significantly different at lower reactor power levels. Thus applying the TS Figure 3.4-1 allowable specific activity for the lower power levels would produce unsatisfactory consequences due to the much greater assumed initial RCS activity level.

The second issue was for the accident induced iodine spiking scenario. The original letdown line break analysis assumed a guillotine break that resulted in a reactor trip and isolation at about 426 seconds. The FSAR Figure 15.1-75 accident induced iodine spiking curve was then applied to the mass release over this time period to determine the activity release to the environment. Since for this event, the RCS activity increases as the transient progresses, the maximum activity release may be for a smaller break size that does not reach reactor trip until operator action is typically credited (30 minutes). In the Waterford 3 revised analysis, it was determined that the largest mass and activity release was for a break size that did not result in a reactor trip and isolation until 30 minutes.

Also, during the re-analysis it was 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 equal to approximately one charging pump. However, during periods of elevated RCS activity levels, 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.

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[3] NUREG-0787, Safety Evaluation Report Related To The Operation Of Waterford Steam Electric Station, Unit 3

3.0 BASIS FOR PROPOSED CHANGE (SAFETY ANALYSES)

The safety analyses were performed as two activities. The first part uses the Nuclear Steam Supply System (NSSS) computer code CESEC-III, an NRC staff approved code as described in the FSAR Section 15.0.3.1.4[4], to determine the transient primary mass release. The second portion is the dose calculation that determines the radiological consequences.

3.1 NSSS Transient Analysis

The transient simulation determined the mass release due to the letdown line break event. The main differences between the original FSAR analysis and the new analysis are listed and described below.

PARAMETER	ORIGINAL ANALYSIS	NEW ANALYSIS
Computer Code	CEFLASH-4AS	CESEC-III
Core Power, Mwt (with RCP Heat)	3478	3480.6
Core Inlet Coolant Temperature, F	557.5	560
RCS Flowrate, lbm/hr	148×10^6	170.2×10^6
RCS Pressure, psia	2250	2300
Trip Credited	CPC DNBR Low Pressurizer Pressure	CPC Hot Leg Saturation Low Pressurizer Pressure
Pressurizer Pressure Control System	None	Automatic
Pressurizer Level Control System	None	Automatic
Letdown Line Modeling	None	Modeled
Limiting Break Size, ft^2	0.01553	0.0094
Isolation Valve Closure Time, sec	5	10

The difference in power level is due to a slight increase in the assumed Reactor Coolant Pump (RCP) heat load into the RCS. The RCS inlet temperature, pressure, and flowrate were chosen based upon limiting parametric studies.

The original analysis was based upon the CEFLASH-4AS computer code. The CEFLASH-4AS code is a NRC staff approved code as described in the FSAR Section 15.0.3.1.11[5]. The CEFLASH-4AS code is primarily used to calculate the hydraulic response of Small Break Loss Of Coolant Accidents (LOCAs). The new analysis used the

[4] Reference Topical Report CENPD-107

[5] Reference Topical Report CENPD-137

CESEC-III computer code which is also an NRC staff approved code as described in the FSAR Section 15.0.3.1.4[6]. The CESEC computer program is a NSSS simulation code.

The original analysis credited the Core Protection Calculator (CPC) low DNBR trip and low pressurizer pressure trip. The CPC low DNBR trip was initiated at the CPC pressure floor which occurs when the pressurizer pressure reaches the CPC lower pressure range limit. The new analysis credits the CPC hot leg saturation trip and the low pressurizer pressure trip. The CPC hot leg saturation trip occurs when the hot leg temperature and pressure indicate that saturation conditions have been reached in the hot legs. Both the original and new analysis credited trips are safety related, redundant, and are considered applicable for the selected trip functions.

The original analysis code did not contain the ability to model a pressurizer pressure control system or a pressurizer level control system. The lack of pressurizer control causes a more rapid depressurization and, consequently, an earlier low pressurizer pressure trip. The shorter transient is non-conservative because it results in less RCS mass release during the simulation. The new analysis models these control systems in automatic mode to delay the time of reactor trip and increase the RCS mass release.

The original analysis neglected the flow resistance in the letdown line between the RCS and outside containment. The new analysis assumed critical flow through the break consistent with the original analysis but accounted for the letdown line losses.

The new analysis credits the CPC Hot Leg Saturation trip resulting in a smaller break being limiting. A larger break causes rapid depressurization resulting in the saturation point of coolant being reached. Thus, the limiting break size is one that extends the length of the transient to approximately 30 minutes and results in the maximum primary coolant and activity release.

The isolation valve stroke time was changed from 5 seconds to 10 seconds and made consistent with the isolation time required in the TRM.

3.2 Radiological Consequences

The dose calculation determined the radiological consequences due to the letdown line break event. The mass release determined from the transient simulation is used as an input. The main differences between the original FSAR analysis and the new analysis are listed and described below.

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[6] Reference Topical Report CENPD-107

PARAMETER	ORIGINAL ANALYSIS	NEW ANALYSIS
Accident Induced Iodine Spike Filtration Flow, gpm	44 (1 pump)	144 (3 pumps)
Dose Conversion Factors I-131 I-132 I-133 I-134 I-135	TID-14844[7] 1 0.036 0.27 0.017 0.084	ICRP-30[8] 1 0.0057 0.1636 0.0010 0.0282
Initial RCS Activity, μCi/gm	6.5	1
Pre-existing Iodine Spike, μCi/gm	50 (SER Analysis)	60
Decay Heat	1973 ANS Standard	1979 ANS Standard

The maximum letdown flow is 128 gpm. The new analysis used 144 gpm to conservatively bound the maximum letdown and identified/unidentified leakage. The maximum values are used because at elevated RCS activities site procedures direct letdown flow to be maximized to clean up the RCS. For the accident induced iodine spiking doses, the 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.

The original analysis Dose Conversion Factors (DCFs) were based upon the TID 14844 information. The new analysis dose conversion factors are based upon ICRP-30 standards. The ICRP-30 standards have been widely accepted and are appropriate for this application. The new analysis uses the ICRP-30 DCFs for the calculations of radiological consequences with respect to 10CFR100 criteria. The new calculation is not intended to pursue alternative source terms at this time.

The original analysis used an initial RCS activity equal to 6.5 μCi/gm which corresponds to 1% failed fuel. The new analysis uses the TS LCO limit of 1 μCi/gm for RCS activity. The use of the less conservative RCS activity value is acceptable based upon the TS requiring the RCS activity be maintained less than or equal to this value. The use of the TS value is consistent with the activity the NRC staff used for their confirmatory analysis.

A pre-existing iodine spike of 60 μCi/gm was assumed, rather than the 50 μCi/gm in the NRC staff SER analysis. This value (60 μCi/gm) is consistent with the current TS 48 hour

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[7] TID 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962.

[8] ICRP (1979), International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, Annals of the ICRP Vol. 2, Nos. 3/4.

Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

action limit at 100% power[9]. This value is also being proposed as the maximum allowable TS 48 hour limit at all power levels.

The decay heat used in the new analysis was updated to include the 1979 ANS Decay Heat Curve with uncertainties (+2 sigma) that include long term actinides. Both the 1973 and 1979 ANS curves have a minimal impact on the dose consequences because the activity release is dominated by the letdown line break and not by the secondary side releases.

3.3 Results

The new analysis corrects the modeling of the letdown isolation function and also models the iodine spiking issue in accordance with the guidelines of the Standard Review Plan 15.1.5, Appendix A. The new analysis results demonstrate that the calculated doses are within the 10CFR100 regulatory requirements.

The original analysis (current FSAR), new analysis, and SER radiological consequences are listed below. The SRP acceptance criteria and 10CFR100 Limits are also listed for comparison purposes.

Exclusion Area Boundary Dose, rem

Event Scenario	**Original Analysis**	**New Analysis**	**SER NRC Staff Results**	**SRP Acceptance Criteria**	**10CFR Part 100 Limit**
Thyroid, no spike	49	5	5.3	30	300
Thyroid, induced spike	140	70	16	30	300
Thyroid, existing spike	**	200	265	**	300
WB*, no spike	0.24	0.3	**	2.5	25
WB*, induced spike	0.33	0.4	**	2.5	25
WB*, existing spike	**	1	**	**	25

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

[9] Reference TS 3/4.4.7 action a., and Figure 3.4-1

Low Population Zone Dose, rem

Event Scenario	FSAR	New Analysis	SER NRC Staff Results	SRP Acceptance Criteria	10CFR Part 100 Limit
Thyroid, no spike	5.5	1	0.1	30	300
Thyroid, induced spike	15	10	0.25	30	300
Thyroid, existing spike	**	25	4.2	**	300
WB*, no spike	0.027	0.05	**	2.5	25
WB*, induced spike	0.036	0.1	**	2.5	25
WB*, existing spike	**	0.2	**	**	25

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

The Exclusion Area Boundary (EAB) thyroid dose reported in the FSAR is 49 rem. This exceeds the SRP acceptance criteria of 30 rem. The NRC staff review, as reported in the Safety Evaluation Report (SER)[10], approved Waterford 3 operation based upon the conclusions of their own event analysis. Those results were based upon an RCS activity of 1 μCi/gm and a pre-existing iodine spike upper limit of 50 μCi /gm that demonstrated acceptable EAB Thyroid results with respect to the SRP.

The new analysis results meet the SRP acceptance criteria with the exception of the EAB accident induced iodine spiking thyroid dose. The SRP acceptance criteria is a small fraction of the 10CFR100 limits (30 rem). The new analysis results are 70 rem which fall well within the 10CFR100 limits of 300 rem.

4.0 JUSTIFICATION

A letdown line break starting from the most limiting parameters allowed by the TS LCO (1μCi/gm) on RCS activity, pressure, temperature, primary to secondary leakage, and proceeding unmitigated for 30 minutes is highly unlikely. The additional use of conservative assumptions such as an iodine spiking factor of 500, maximum bounding letdown flow, worst case 95 percentile atmospheric dispersion factors, flashing fraction based on 560°F even though the break flow would travel through the regenerative heat exchanger and cool down, no activity plate out, no ground deposition, and no activity decay in the transit to the exclusion area boundary significantly increases the overall conservative nature of the calculation.

The table below lists some of the conservative assumptions used in the new analysis in comparison to the typical Waterford 3 normal operating experience.

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[10] NUREG-0787, Safety Evaluation Report Related To The Operation Of Waterford Steam Electric Station, Unit 3, Section 15.4.3

Parameter	**New Analyses**	**Normal Plant Operation**
Power (Mwt)	3457.8	3390
Cold Leg Temperature (°F)	560	545
Pressurizer Pressure (psia)	2300	2250
RCS Flow (% of Design)	115	107
Limiting Break Size (ft^2)	0.0094	N/A
RCS Steady State Activity (μCi/gm DEQ I-131)	1	<0.1
RCS Pre-Existing Spike Activity (μCi/gm DEQ I-131)	60	N/A
Accident Induced Iodine Spiking Factor	500	<200 (trip)
Letdown/Charging Flow (gpm)	144	44
Atmospheric Dispersion Factors, X/Q (sec/m^3)	$6.3x10^{-4}$ (5%)	$5.5x10^{-5}$ (50%)
Primary to Secondary Leakage (gpm) Total	1	<Detectable
SG Steady State Activity (μCi/gm DEQ I-131)	0.1	<Detectable
Operator Action, min	30	10-20

Currently, FSAR Table 15.6-4 lists the 'Realistic' EAB thyroid dose as 0.46 rem. The realistic dose is based upon no iodine spike, 50 percentile X/Q, and 0.12% failed fuel RCS activity. The best estimate dose consequences using the new analysis methodology with the normal plant operating parameters listed above would remain below 0.46 rem even for the accident induced iodine spiking event.

The new analysis accident induced iodine spiking results would remain below the SRP acceptance criteria if any one of the following normal plant operating parameters were used: RCS steady state activity, iodine spiking factor, letdown flow, or atmospheric dispersion factors. From the table above, the normal plant operating parameters demonstrate the values used in the analysis are conservative by a factor of 10 for RCS steady state activity, a factor of 2.5 for iodine spiking factor, about a factor of 3 for letdown flow, and about a factor of 11 for the atmospheric dispersion factors.

Therefore, Waterford 3 proposes that due to the low occurrence probability of this event at the bounding conditions plus the conservative nature of the calculation that it is reasonable and appropriate to allow the accident induced iodine spiking consequences to fall well within the 10CFR100 limits (25% of Part 100 or 75 rem).

5.0 DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Entergy Operations, Inc. is proposing that the 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 Operations, Inc. 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 FSAR change 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 proposed FSAR change meets the original SER acceptance criteria with the exception of the Exclusion Area Boundary (EAB) accident induced iodine spiking thyroid dose. The SRP acceptance criteria for the EAB accident induced iodine spiking thyroid dose is a small fraction of the 10CFR100 limits (30 rem). The proposed change falls well within 10CFR100 limits (75 rem).

The EAB accident induced iodine spiking thyroid dose consequences are considered acceptable and reasonable for the following reasons.

- The letdown line break event starting from the most limiting parameters allowed by the TS LCO on RCS activity, pressure, temperature, primary to secondary leakage, and proceeding unmitigated for 30 minutes is highly unlikely. The additional use of conservative assumptions such as an iodine spiking factor of 500, maximum bounding letdown flow, worst case 95 percentile atmospheric dispersion factors, flashing fraction based on 560 °F even though the break flow would travel through the regenerative heat exchanger and cool down, no activity plate out, no ground deposition, and no activity decay in the transit to the exclusion area boundary significantly increases the overall conservative nature of the calculation.
- Currently, FSAR Table 15.6-4 lists the 'Realistic' EAB thyroid dose as 0.46 rem. The realistic dose is based upon no iodine spike, 50 percentile X/Q, and 0.12% failed fuel RCS activity. The best estimate dose consequences using the new analysis methodology with the normal plant operating parameters would remain below 0.46 rem even for the accident induced iodine spiking event.
- The new analysis accident induced iodine spiking results would remain below the SRP acceptance criteria if any one of the following normal plant operating parameters were used: RCS steady state activity, iodine spiking factor, letdown flow, or atmospheric dispersion factors.

The letdown line break consequences are considered acceptable due to the unlikeliness of the event and conservative nature of the analyses. The 'no iodine spike' results remain within a small fraction of the 10CFR100 limits; the 'accident induced iodine spike' results fall well within the 10CFR100 limits; and the 'pre-existing iodine spike' results are within 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 pre-existing iodine spike is not an accident initiator and the FSAR change does not affect any plant Structure, Systems, or Component (SSC) 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 proposed FSAR change meets the original SRP acceptance criteria with the exception of the Exclusion Area Boundary (EAB) accident induced iodine spiking thyroid dose. The SRP acceptance criteria for the EAB accident induced iodine spiking thyroid dose is a small fraction of the 10CFR100 limits (30 rem). The proposed change falls well within 10CFR100 limits (75 rem).

The EAB accident induced iodine spiking thyroid dose consequences are considered not to be a significant reduction in the margin of safety for the following reasons.

- The letdown line break event starting from the TS LCO on RCS activity, pressure, temperature, primary to secondary leakage, and proceeding unmitigated for 30 minutes is highly unlikely. The additional use of conservative assumptions such as an iodine spiking factor of 500, maximum bounding letdown flow, worst case 95 percentile atmospheric dispersion factors, flashing fraction based on 560 °F even though the break flow would travel through the regenerative heat exchanger and cool down, no activity plate out, no ground deposition, and no activity decay in the transit to the exclusion area boundary significantly increases the overall conservative nature of the calculation.
- The FSAR Table 15.6-4 lists the 'Realistic' EAB thyroid dose as 0.46 rem. The realistic dose is based upon no iodine spike, 50 percentile X/Q, and 0.12% failed fuel RCS activity. The best estimate dose consequences using the new analysis methodology with the normal plant operating parameters would remain below 0.46 rem even for the accident induced iodine spiking event.
- The new analysis accident induced iodine spiking results would remain below the SRP acceptance criteria if any one of the following normal plant operating parameters were used: RCS steady state activity, iodine spiking factor, letdown flow, or atmospheric dispersion factors.

The letdown line break consequences are considered acceptable due to the unlikeliness of the event and conservative nature of the analyses. The 'no iodine spike' results remain within a small fraction of the 10CFR100 limits; the 'accident induced iodine spike' results fall well within the 10CFR100 limits; and the 'pre-existing iodine spike' results are within the 10CFR100 limits.

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

6.0 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 TS change is conservative in that it will reduce the accident consequences for events occurring at lower power levels. The FSAR change 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 FSAR 'Realistic' results.

ATTACHMENT 2

TO

W3F1-2001-0088

PROPOSED MARKED-UP TECHNICAL SPECIFICATION

IN THE MATTER OF AMENDING

LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

INDEX

LIST OF FIGURES

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LIST OF TABLES (Continued)

REACTOR COOLANT SYSTEM

3/4.4.7 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/Ē microcuries/gram.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2, and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding 60 microcuries/gram DOSE EQUIVALENT I-131 ~~the limit line shown in Figure 3.4-1~~, be in at least HOT STANDBY with Tavg less than 500 °F within 6 hours.

b. With the specific activity of the primary coolant greater than 100/Ē microcuries/gram, be in at least HOT STANDBY with Tavg less than 500 °F within 6 hours.

MODES 1, 2, 3, 4, and 5:

c. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 or greater than 100/Ē microcuries/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

*With T_{avg} greater than or equal to 500 °F.

~~REACTOR COOLANT SYSTEM~~

~~LIMITING CONDITION FOR OPERATION (Continued)~~

~~ACTION: (Continued)~~

~~MODES 1, 2, 3, 4, and 5:~~

~~c. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 or greater than 100/Ē microcuries/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.~~

Moved to page 3/4 4-24

Moved Table 4.4-4 to page 3/4 4-25

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for Ē Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram, DOSE EQUIVALENT I-131 or 100/Ē μCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 % of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

Until the specific activity of the primary coolant system is restored within its limits.

<u>Pages 3/4 4-26 and 3/4 4-27 have been deleted</u>

Next Page is 3/4 4-28

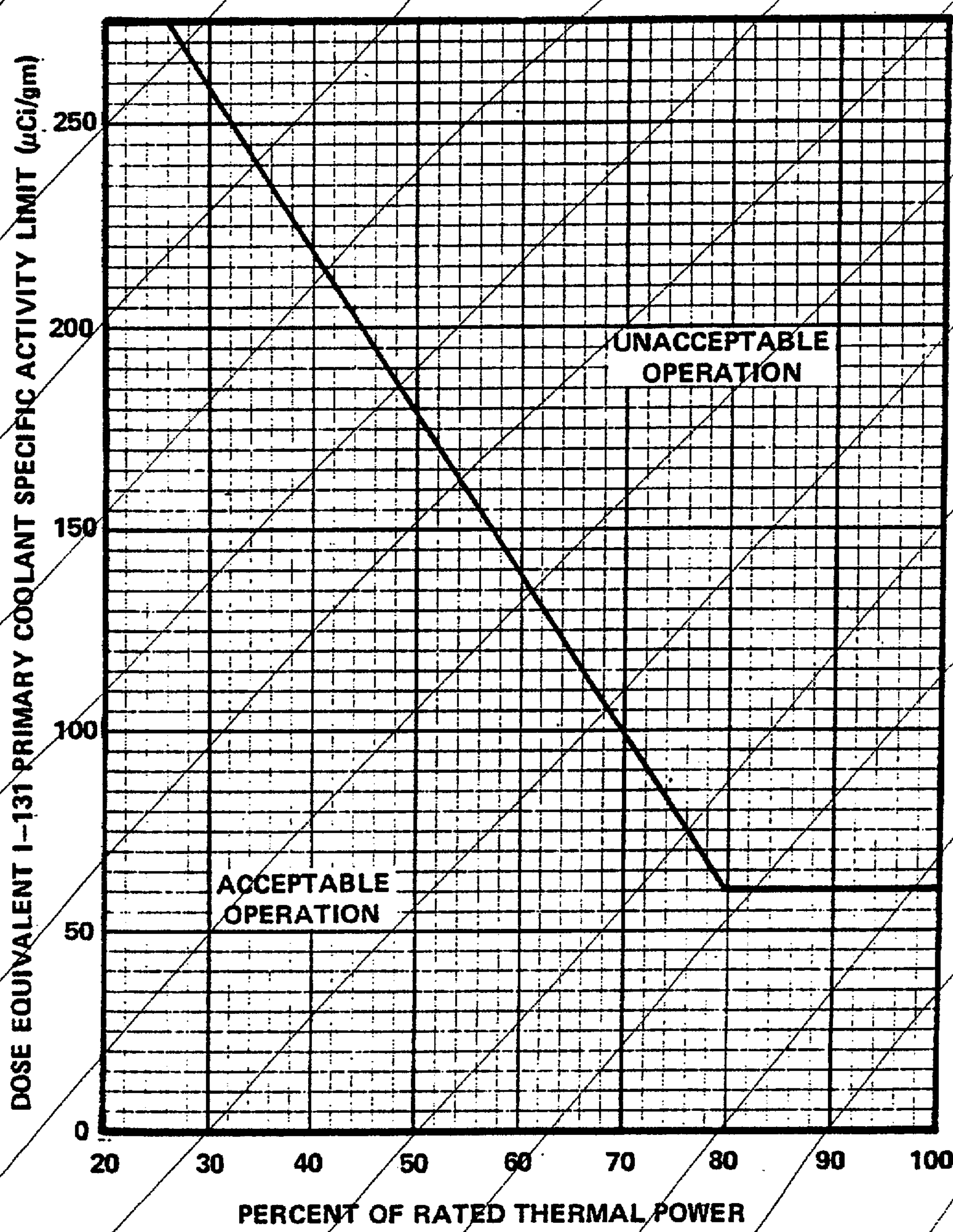


FIGURE 3.4-1

DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY > 1.0 µCi/GRAM DOSE EQUIVALENT I-131

ATTACHMENT 3

TO

W3F1-2001-0088

PROPOSED MARKED-UP UPDATE FINAL SAFETY ANALYSIS REPORT

IN THE MATTER OF AMENDING

LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

LIST OF FIGURES (Cont'd)

15.6 DECREASE IN REACTOR COOLANT SYSTEM INVENTORY

15.6.1 MODERATE FREQUENCY INCIDENTS

There are no moderate frequency incidents resulting from a decrease in reactor coolant inventory.

15.6.2 INFREQUENT INCIDENTS

There are no infrequent incidents resulting from a decrease in reactor coolant inventory.

15.6.3 LIMITING FAULTS

15.6.3.1 Primary Sample or Instrument Line Break

15.6.3.1.1 Identification of Causes and Frequency Classification

The estimated frequency of a primary sample or instrument line rupture classifies it as a limiting fault incident as defined in Reference 1 of Section 15.0. A primary sample or instrument line break provides a release path for reactor coolant outside containment. The line break selected for analysis is the letdown line (two inch Schedule 160 Pipe) which penetrates the containment. This is the largest penetration whose failure could result in an event in this category. ~~This failure would result in larger releases than would be the case for the smaller instrument and sample lines.~~ The break size was investigated to determine the maximum RCS mass release outside containment.

15.6.3.1.2 Sequence of Events and Systems Operation

The integrity of lines containing primary coolant external to the containment is significant radiologically since a rupture of this barrier results in the release of reactor coolant outside containment. Following such a break, the RCS pressure decreases due to the loss of reactor coolant. ~~When pressurizer pressure has decreased to the low pressure floor in the CPC's, a reactor trip is initiated on low DNBR.~~ The pressure will decrease to the point where the reactor coolant volume will become saturated. Once the maximum calculated CPC hot leg temperature exceeds the saturation temperature, a reactor trip is initiated on CPC hot leg saturation. The safety injection actuation signal (SIAS) on low pressurizer pressure terminates the break flow by isolating the letdown line inside containment, and the reactor coolant inventory is replenished by the safety injection pumps and by the charging pumps.

Operation of the HPSI pumps as well as the charging pumps ensures that the core will not uncover and prevents any significant increase in clad temperatures.

Table 15.6-1 shows the sequence of events following a letdown line break.

15.6.3.1.3 Core and System Performance

15.6.3.1.3.1 Mathematical Model

The NSSS response to a letdown line break was simulated using the ~~CEFLASH-4AS blowdown code described in Reference 1, Supplement 1~~ CESEC-III computer program described in Section 15.0.

15.6.3.1.3.2 Input Parameters and Initial Conditions

The initial conditions and input parameters of the NSSS assumed in the analysis are listed in Table 15.6-2.

15.6.3.1.3.3 Results

The response of the NSSS following a letdown line break begins with a decrease in pressurizer level and pressure. The decreasing pressure transient is shown in Figure 15.6-1. At approximately ~~406~~1780 seconds after the break, ~~pressurizer pressure has dropped to the CPC low pressure floor and a reactor trip is initiated on low DNBR~~ RCS pressure has decreased such that the calculated CPC saturation temperature is exceeded by the hot leg temperature and a reactor trip is initiated. The turbine trip on reactor trip results in an increase in secondary side pressure to the steam generator safety valve set pressure. An SIAS is also generated on low pressurizer pressure, isolating the letdown line and terminating the break flow at about ~~426~~1800 seconds.

The reactor coolant inventory is replenished by the HPSI pumps and by the charging pumps. Operation of these pumps ensures that the core will not uncover and prevents any significant increase in clad temperature.

After 30 minutes, the operator is assumed to start a plant cooldown.

15.6.3.1.4 Barrier Performance

15.6.3.1.4.1 Mathematical Model

The mathematical model used for evaluation of Barrier Performance is described in Subsection 15.6.3.1.3.

15.6.3.1.4.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used for evaluation of Barrier Performance are the same as those described in Subsection 15.6.3.1.3.

15.6.3.1.4.3 Results

At about ~~426~~1800 seconds into the transient, the ruptured line is isolated, terminating the leak flow. Prior to isolation of the line, ~~126,169~~less than 67,000 pounds of primary coolant have been released from the RCS.

15.6.3.1.5 Radiological Consequences

15.6.3.1.5.1 Design Basis, No Iodine Spike

15.6.3.1.5.1.1 Physical Model

A break in fluid-bearing lines which penetrate the containment could result in the release of radioactivity to the environment. There are no instrument lines connected to the RCS which penetrate the containment. There are, however, other piping lines from the RCS to the Chemical and Volume Control System (CVCS) and the Process Sampling System which penetrate the containment. The most severe rupture with respect to radioactivity release during normal plant operation is the rupture of the letdown line outside containment. For such a break, the reactor coolant letdown flow would have passed from the cold leg and through the regenerative heat exchanger.

It was calculated that about ~~426~~1800 seconds would elapse before an SIAS is initiated on low pressurizer pressure and the letdown line isolation valves are shut. The reactor coolant mass released to the Reactor Auxiliary Building (RAB) is ~~about 126,169~~less than 67,000 lbm.

15.6.3.1.5.2 Assumptions and Parameters

The major assumptions and parameters used in the analysis are listed in Table 15.6-2 and 15.6-3 are discussed below:

a) The reactor coolant equilibrium activity is ~~based on long-term operation at 105 percent of the ultimate core power level of 3390 Mwt with 1.0 percent failed fuel. The activities are given in Table 11.1-2~~ the Technical Specification limit of 1.0 μCi/gm Dose Equivalent (DEQ) Iodine 131 (I-131).

b) ~~A total of 126,169~~Less than 67,000 pounds of reactor coolant are spilled.

c) All the noble gases in spilled reactor coolant are released to the atmosphere.

d) The fraction of water flashing to steam was calculated and was defined as the fraction of iodines in the water that volatilize. The reactor coolant temperature was assumed to be ~~550~~560°F. The fraction of iodines calculated to volatilize was 0.4.

e) No credit is taken for mixing or holdup of the activity released to the RAB atmosphere.

f) The activity released from the ruptured letdown line is assumed to be released directly to the environment during the two-hour period immediately following the accident.

g) No credit is taken for ground deposition or decay in transit to the exclusion area boundary or outer boundary of the low population zone (LPZ).

15.6.3.1.5.1.3 Mathematical Model

Models used in the analysis are described in the following sections:

a) The meteorological conditions assumed present during the course of the accident are based on X/Q values which are expected to be conservative 95 percent of the time.

Calculational methods for X/Qs are presented in Subsection 2.3.4. For the design basis accident, five percent level X/Qs were used (Table 2.3-136).

b) The potential thyroid inhalation dose and total-body gamma immersion dose to an individual exposed at the exclusion area boundary or LPZ outer boundary are obtained using the models given in equations 19 and 20 of Appendix 15B.

15.6.3.1.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The reactor coolant spilled in the Reactor Auxiliary Building (RAB) is collected in the floor drain sumps. From there, it is pumped to the radwaste treatment system. Thereafter, the only release paths that present a radiological hazard involve the volatile fraction of spilled coolant.

15.6.3.1.5.1.5 Uncertainties and Conservatisms in the Evaluation of the Results

The principal uncertainties and conservatisms in the calculation of the resultant doses following a letdown line rupture arise from the unknown extent or reactor coolant contamination by radionuclides, the quantity of coolant spilled, the fraction of radionuclides that volatilize, the fraction of the spilled activity that escapes the RAB, and the meteorological conditions at the time of the accident. Each of these uncertainties is treated by taking worst-case or conservative assumptions.

a) Reactor coolant equilibrium activities are based on ~~1.0 percent failed fuel~~ the Technical Specification limit, which is ~~a factor of two to eight~~ greater than that normally observed in past PWR operation.

b) The quantity of coolant spilled is maximized by ~~assuming critical flow through the letdown line~~ determining the break size that produces the largest mass release.

c) The fraction of iodines calculated to volatilize is based on a reactor coolant temperature of ~~550~~560°F. This temperature does not take credit for cooling provided by the regenerative heat exchanger. The resulting fraction of iodines released (40 percent) would decrease as the coolant temperature decreased.

d) No credit is taken for the effects of ~~in~~retention of radioactivity (plate out) which could occur within the RAB, reducing the amount of activity released to the environment.

e) The meteorological conditions assumed during the course of the accident are based on X/Q values which are expected to be conservative 95 percent of the time. This condition results in the poorest values or atmospheric dispersion calculated for the exclusion area boundary or LPZ outer boundary. Further, no credit is taken for the transit time of activity from the point of release to the exclusion area boundary or LPZ outer boundary. Hence, the radiological consequences evaluated under these conditions are conservative.

15.6.3.1.5.1.6 Results

Offsite Doses

The radiological consequences resulting from a letdown line rupture have been conservatively calculated using assumptions and models described above. The thyroid inhalation dose and total-body gamma immersion dose have been calculated for the initial 2 hour period at the exclusion area boundary and for the duration of the accident at the LPZ outer boundary.

The results are listed in Table 15.6-4. The resultant doses are small fractions of the guidelines of 10CFR100 without the iodine spike ~~and within the guidelines of 10CFR1000 with the iodine spike~~.

15.6.3.1.5.2 Design Basis, Iodine Spike Caused by the Accident

In this evaluation, the radiological consequences of the letdown line rupture was evaluated assuming that the accident causes an iodine spike. The mathematical models, assumptions, and parameters used in this analysis are identical with the design basis evaluation without an iodine spike discussed in Subsection 15.6.3.1.5.1 with the following exception.

At the initiation of the letdown line break, the I-131 equivalent source term is assumed to increase as shown in Figure 15.1-75a. This figure is based on a factor of 500 increase in the iodine release rate.

The activity released is presented in Table 15.6-3. Radiological consequences are presented in Table 15.6-4 and are within the guidelines of 10CFR100.

~~15.6.3.1.5.3 Realistic Analysis~~

~~A realistic analysis of a small line break outside of containment was performed. The realistic analysis was identical with the design basis evaluation (no iodine spike) presented in Subsection 15.6.3.1.5.1 with the following exceptions:~~

~~a) The reactor coolant system equilibrium activity prior to the accident was assumed to be based on 3390 Mwt and 0.12 percent failed fuel.~~

~~b) The atmospheric dispersion factor (X/Q) used were the 50 percent level factors presented in Table 2.3-136.~~

~~Assumptions and parameters are presented in Table 15.6-3. Results are presented in Table 15.6.4.~~

15.6.3.2 Steam Generator Tube Rupture

Three cases of a Steam Generator Tube Rupture (SGTR) are included:

a. Steam Generator Tube Rupture (without a concurrent Loss of Offsite Power) (Subsection 15.6.3.2.1)

b. Steam Generator Tube Rupture analysis with concurrent Loss of Offsite Power which demonstrates that DNB does not occur (DNBR Performance Case, Subsection 15.6.3.2.2), and

c. The analysis of SGTR with concurrent Loss of Offsite Power for radiological consequences, which accounts for the impact of potential uncovery of the rupture during the event (Radiological Consequences Case, Subsection 15.6.3.2.3).

TABLE 15.6-1

SEQUENCE OF EVENTS FOR A LETDOWN LINE BREAK OUTSIDE CONTAINMENT

Time (seconds)	Event	Setpoint or Value
0.0	Line rupture occurs	---
~~406.2~~1778.8	~~Low DNBR trip, psia~~CPC Hot Leg Saturation trip	~~1,728~~---
~~407.1~~1779.4	CEAs begin to drop into core	---
~~410.7~~1782.4	CEAs 90 percent inserted	---
~~421.3~~1788.8	Safety injection actuation signal, psia	1,560
~~426.3~~1798.8	Isolation of ruptured letdown line	---
~~452.0~~1800.0	SIS flow initiated	---
1800.0	Operator initiated plant cooldown	---
9000.0	Shutdown cooling initiated, F	350

TABLE 15.6-2

ASSUMPTIONS FOR LETDOWN LINE BREAK OUTSIDE CONTAINMENT

Parameter	Assumption
Initial core power, Mwt	~~3478~~ 3457.8
Core inlet coolant temperature, °F	~~557.5~~ 560.0
Core outlet coolant temperature, °F	~~616.6~~ 611.7
Initial RCS flowrate, lbm/hr	~~148×10^6~~ 170.2×10^6
Initial RCS pressure, psia	~~2250.0~~ 2300.0
Steam generator secondary pressure, psia	~~998.0~~ 961.4
Secondary relief valve setpoint (lowest bank), psia	~~1180.0~~ 1117.6
Moderator temperature coefficient, 10^{-4} Δp°F	~~+0.5~~ -3.3

See new Table 15.6-3 (Sheet 1 of 2)

TABLE 15.6-3 (Sheet 1 of 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LETDOWN LINE RUPTURE

Parameter	Design Bases Assumptions	Realistic Assumptions
Source Data:		
A. Core Power Level, MWt	3,560	3,390
B. Fraction failed fuel, percent	1	0.12
C. Assumed decay time, seconds	0	0
D. Reactor coolant equilibrium activities	Table 11.1-2	Table 11.1-3
E. Mass of coolant released, lb	126,169	126,169
F. Duration of spill, seconds	426	426
Activity Release Data:		
A. Release assumptions		
Fraction of noble gases immediately released, %	100	100
Fraction of iodines immediately released, %	40	40
Activity released to atmosphere, Ci		
No iodine spike		
Isotope		
I-131	1.07(2)	1.16(1)
I-132	3.02(1)	3.07(0)
I-133	1.35(2)	1.42(1)
I-134	1.31(1)	1.40(0)
I-135	5.92(1)	6.32(0)
Xe-131m	1.37(2)	8.77(0)
Xe-133	1.91(4)	1.37(3)
Xe-135m	6.30(1)	9.40(-1)
Xe-135	5.29(2)	2.54(1)
Xe-138	3.20(1)	3.19(0)

See new Table 15.6-3 (Sheet 2 of 2)

TABLE 15.6-3 (Sheet 2 of 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LETDOWN LINE RUPTURE

Parameter	Design Bases Assumptions	Realistic Assumptions
Isotope		
Kr-85	2.94(2)	1.66(1)
Kr-87	7.21(1)	4.34(0)
Kr-88	2.34(2)	1.45(1)
With iodine spike		
Isotope		
I-131	2.96(2)	-
I-132	8.36(2)	-
I-133	3.74(2)	-
I-134	3.63(1)	-
I-135	1.64(2)	-
Xe-131m	1.37(2)	-
Xe-133	1.91(4)	-
Xe-135m	6.30(1)	-
Xe-135	5.29(2)	-
Xe-138	3.20(1)	-
Kr-85	2.94(2)	-
Kr-87	7.21(1)	-
Kr-88	2.34(2)	-
Dose Calculation Methodology Dispersion Data	Appendix 15B	Appendix 15B
Atmospheric dispersion factors, sec/m^3	5% level χ/Qs (Table 2.3-136)	50% level χ/Qs (Table 2.3-136)

TABLE 15.6-3 (Sheet 1 OF 2)

PARAMETER USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LETDOWN LINE RUPTURE

Parameter	Design Bases Assumptions
1. Source Data:	
A. Core Power Level, Mwt	3559.5
B. Assumed decay time, seconds	0
C. Reactor coolant equilibrium Activities, μCi/gm DEQ I-131	1
D. Mass of coolant released, lb	67,000
E. Duration of spill, seconds	1800
2. Activity Release Data:	
A. Fraction of noble gases immediately released, %	100
B. Fraction of iodines immediately released, %	40
C. Total EAB Iodine Activity released to atmosphere, No iodine spike, Ci DEQ I-131	16
D. Total EAB Iodine Activity released to atmosphere, Accident induced iodine spike, Ci DEQ I-131	260
E. Total EAB Noble Gas Activity released to atmosphere, No iodine spike, Ci	2750
F. Total EAB Noble Gas Activity released to atmosphere, Accident induced iodine spike, Ci	2750

TABLE 15.6-3 (Sheet 2 OF 2)

PARAMETER USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LETDOWN LINE RUPTURE

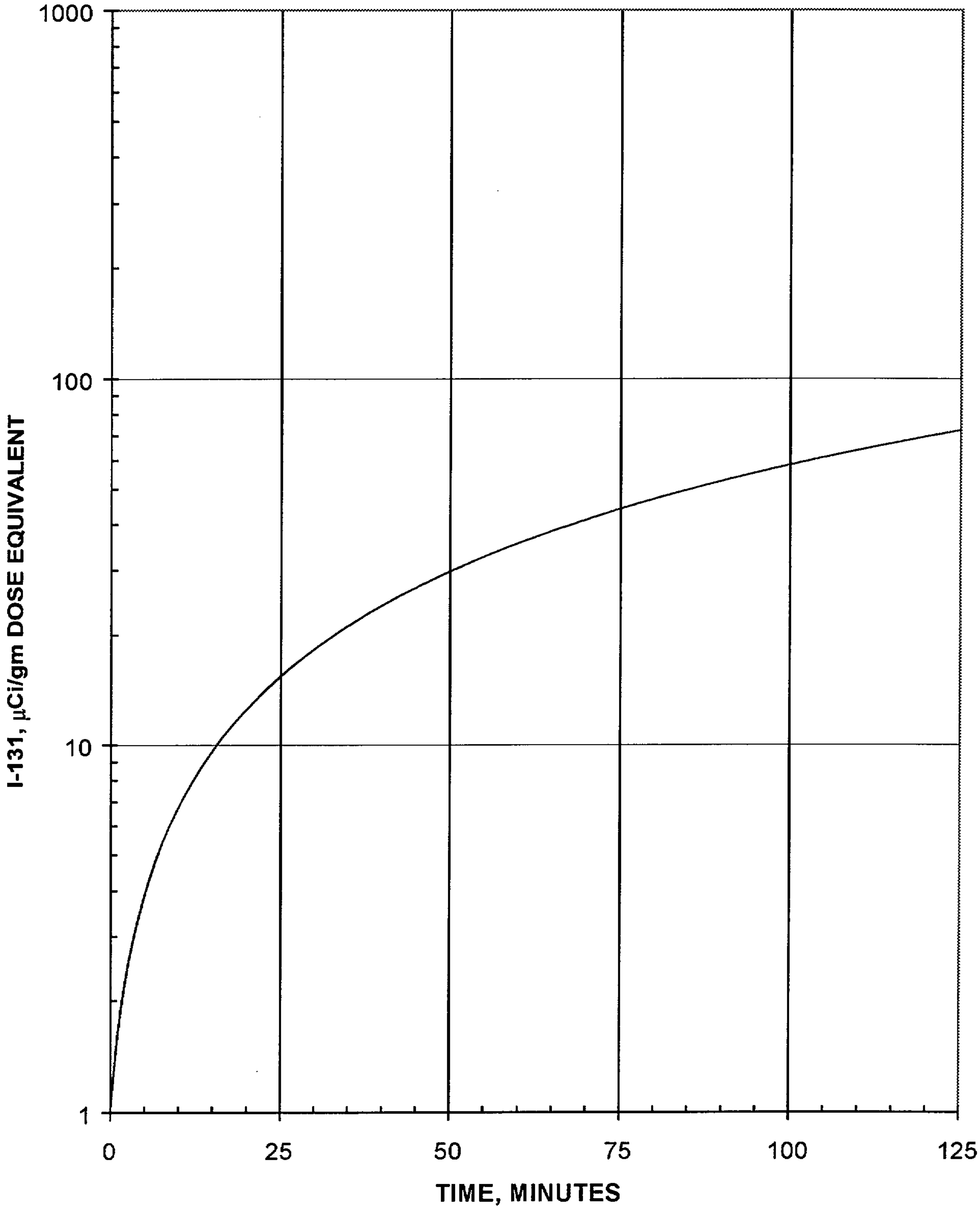
Parameter	Design Bases Assumptions
G. Total LPZ Iodine Activity released to atmosphere, No iodine spike, Ci DEQ I-131	20
H. Total LPZ Iodine Activity released to atmosphere, Accident induced iodine spike, Ci DEQ I-131	335
I. Total LPZ Noble Gas Activity released to atmosphere, No iodine spike, Ci	2860
J. Total LPZ Noble Gas Activity released to atmosphere, Accident induced iodine spike, Ci	2860
K. Dose Calculation Methodology Dispersion Data	Appendix 15B
L. Atmospheric dispersion factors, sec/m^3	5% level X/Qs (Table 2.3-136)

TABLE 15.6-4

RADIOLOGICAL CONSEQUENCES OF A LETDOWN LINE RUPTURE IN THE REACTOR AUXILIARY BUILDING

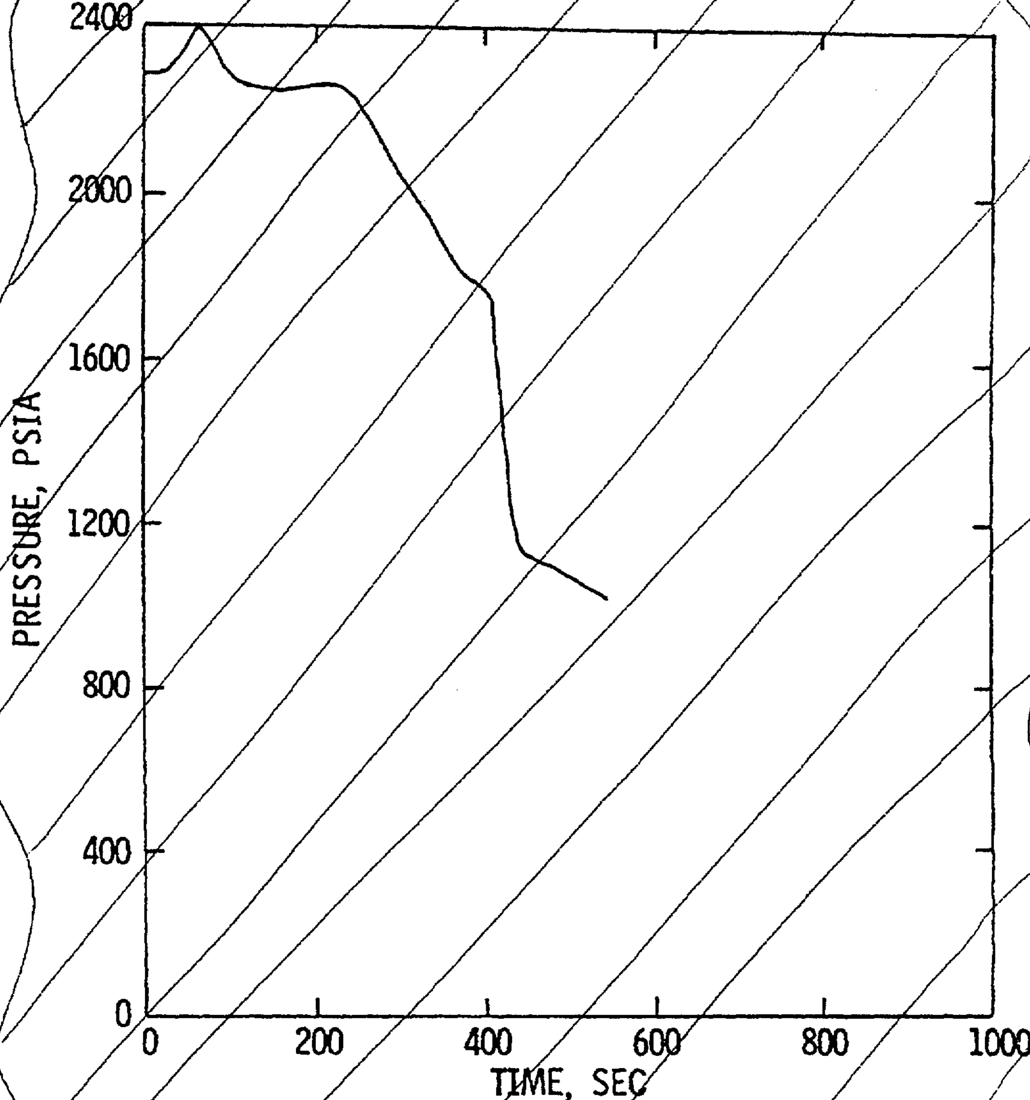
Result	Design Basis Value: No Iodine Spike	Design Basis Value: With Iodine Spike	~~Realistic Value~~
Exclusion area boundary Dose, rem (0-2 hour)			
Thyroid	~~4.9(1)~~ 5	~~1.4(2)~~ 70	~~4.6(1)~~
Whole-body	~~2.4(-1)~~ 0.3	~~3.3(-1)~~ 0.4	~~1.5(-3)~~
LPZ outer boundary dose rem (duration)			
Thyroid	~~5.5(0)~~ 1	~~1.5(1)~~ 10	~~3.5(-2)~~
Whole-body	~~2.7(-2)~~ 0.05	~~3.6(-2)~~ 0.1	~~1.2(-4)~~

New Figure



Waterford 3 Steam Electric Station	REACTOR COOLANT SYSTEM DOSE EQUIVALENT IODINE CONCENTRATION VS. TIME (SPIKING FACTOR = 500)	Figure 15.1-75a

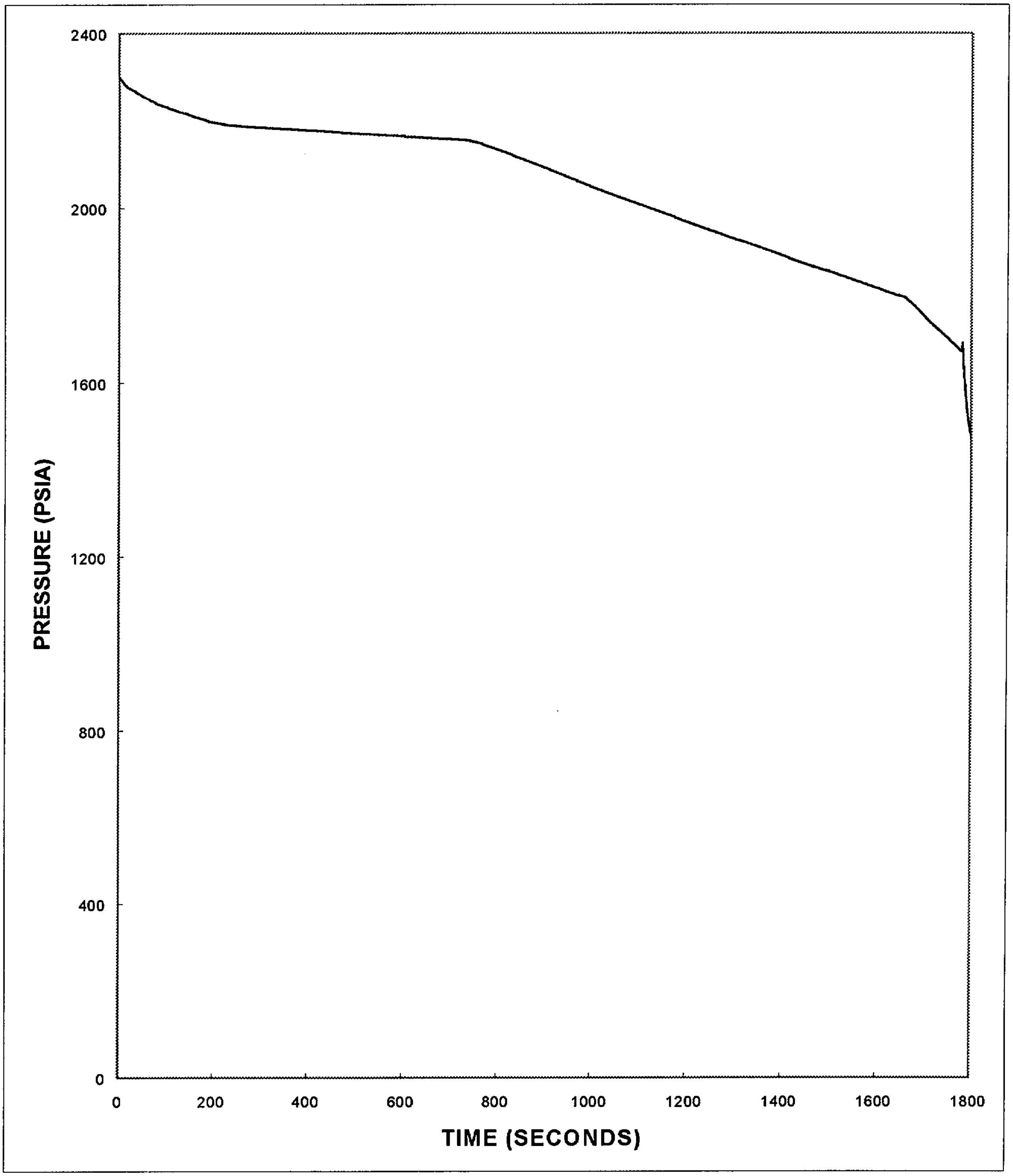
Replace with new figure 15.6-1



REVISION 3 (12/89)

LOUISIANA POWER & LIGHT CO. Waterford Steam Electric Station #3	REACTOR COOLANT SYSTEM PRESSURE VS TIME, LETDOWN LINE BREAK	FIGURE 15.6-1

New Figure 15.6-1



Waterford 3 Steam Electric Station	REACTOR COOLANT SYSTEM PRESSURE VS TIME LETDOWN LINE BREAK	Figure 15.6-1