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SOUTHERN COMPANY
Energy to Serve Your World℠

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant
Final Safety Analysis Report Change Request
for Original Steam Generators on Unit 2
Iodine Spiking due to Increased Letdown Flow Rate

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) proposes to revise the Farley Nuclear Plant (FNP) Updated Final Safety Analysis Report (UFSAR) Chapters 11 and 15. The revision incorporates changes due to revisions to Dose Equivalent Iodine (DEI) analyses. Attachments 1 through 3 contain the submittal information for the Original Steam Generators (OSGs) on Unit 2. Attachment 1 contains the safety analysis explaining the technical basis for the change, Attachment 2 contains the 10 CFR 50.92 Significant Hazards Evaluation, and Attachment 3 contains markups of the affected UFSAR pages. Additional issues identified in Westinghouse Nuclear Safety Advisory Letter NSAL-00-004, "Nonconservatisms in Iodine Spiking Calculations," are also discussed. The proposed changes to UFSAR Chapters 11 and 15 are provided for your review and approval. Upon approval, these changes will be incorporated into the appropriate UFSAR revision.

By NRC letter dated May 10, 1999, Amendment Nos. 142 and 134 were issued for Farley Nuclear Plant (FNP) Units 1 and 2, respectively. These amendments changed the DEI limits from 0.15 μCi/gram to 0.30 μCurie/gram. In the SNC submittal and the associated NRC Safety Evaluation (SE), reference was made to the allowable primary-to-secondary leakage associated with a main steamline break (MSLB) accident. The MSLB accident is the limiting accident due to steam generator alternate repair criteria (ARC) being in place. The change in DEI limits had an associated change in the allowable primary-to-secondary leakage rate from 23.8 gpm to 11.8 gpm.

On February 23, 2000, FNP voluntarily submitted FNP Unit 1 and 2 Licensee Event Report (LER) No. 2000-001. This LER documented that the limiting main steam line break (MSLB) offsite dose analysis for FNP Units 1 and 2 had assumed a non-conservative value for letdown flow rate which affected the iodine source term (i.e., the DEI) used in the analysis. This condition was discovered due to evaluation of a similar condition discovered at Beaver Valley (documented in Beaver Valley LER 50/334 99-002-00). In LER 2000-001, it was stated that SNC would review the recommendations of a Westinghouse NSAL for applicability and further corrective actions as appropriate when the NSAL was issued. NSAL-00-004 was issued on March 7, 2000 regarding this issue.

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SNC is taking the following steps to address this issue:

Impacted dose analyses were re-performed and shielding analyses were evaluated for the bounding letdown flow and Technical Specification DEI for the OSGs currently installed on Unit 2. The existing Farley dose analyses assume a design flow of 60 gpm while plant procedures permit operation with normal letdown of approximately 130 gpm. An uncertainty calculation was performed for the alarm, main control board indicator, and computer point associated with the letdown flow element. Based on these results, an analytical limit of 145 gpm letdown was selected. The letdown flow rate assumed accommodates higher letdown plus flow measurement uncertainty as recommended in Westinghouse NSAL-00-004.

The OSGs will be replaced on Unit 2 in the Spring of 2001. The current projected end of cycle SLB leakage for Unit 2 is 2.2 gpm. This value is well below the current limit of 11.8 gpm and demonstrates that existing margin can be recovered without impacting offsite dose analysis. For Unit 2 OSGs, allowable steam generator tube ARC primary-to-secondary leakage assumed in the MSLB analysis was reduced to offset the increase in assumed letdown at the Technical Specification DEI limit of 0.3 μCi/gm. A reduction in allowable ARC leakage from 11.8 gpm to 5 gpm was required to offset the 145 gpm letdown flow. The new ARC leakage limit was determined as the value necessary to ensure revised MSLB doses at maximum letdown do not exceed those previously reported to the NRC. SGTR doses increased slightly but continue to be bounded by MSLB. Doing this will keep the current Unit 2 Technical Specification limit for DEI (0.3 μCi/gm) conservative and bounding. Therefore, no changes to the Technical Specifications are required.

Sufficient margins in source terms were available to offset any adverse impacts to analyses for shielding. Additionally, source terms for calculated normal releases experienced a reduction at increased letdown such that the doses at the original letdown flow rate of 60 gpm remain bounding.

SNC has reviewed the proposed amendment pursuant to 10 CFR 50.92 and determined that it does not involve a significant hazards consideration. In addition, there is no significant increase in the amounts of effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Consequently, the proposed amendment satisfies the criteria of 10 CFR 51.22 for categorical exclusion from the requirements for an environmental assessment and the human environment is not affected by this amendment. A copy of the proposed changes has been sent to Dr. D. E. Williamson, the Alabama State Designee, in accordance with 10 CFR 50.91(b)(1).

There are two dose analyses of record for Farley; one related to ARC for the OSGs and one related to the Replacement Steam Generators (RSGs). This letter addresses FSAR changes related to the OSGs. SNC letter NEL-00-0170 addresses FSAR changes related to the RSGs.

If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

Dave Morey

Dave Morey

Sworn to and subscribed before me this 25th *day of* August 2000

Martha Gayle Dow

Notary Public

My Commission Expires: November 1, 2001

WAS/maf:Original DEI NRC Letter.doc

Attachments:
1. Safety Analysis
2. Significant Hazards Evaluation
3. Markups of Affected UFSAR Pages

cc: Southern Nuclear Operating Company
Mr. L. M. Stinson, General Manager - Farley

U. S. Nuclear Regulatory Commission, Washington, D. C.
Mr. L. M. Padovan, Licensing Project Manager - Farley

U. S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. T. P. Johnson, Senior Resident Inspector – Farley

Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer

Attachment 1

Joseph M. Farley Nuclear Plant
Final Safety Analysis Report Change Request
for Original Steam Generators on Unit 2
Iodine Spiking due to Increased Letdown Flow Rate

Safety Analysis

Joseph M. Farley Nuclear Plant
Final Safety Analysis Report Change Request
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Iodine Spiking due to Increased Letdown Flow Rate

Safety Analysis

The proposed license amendment request is to revise UFSAR described offsite dose analyses based on changes to the letdown flow rate and iodine spike postulated concurrent with a Main Steam Line Break (MSLB) or a Steam Generator Tube Rupture (SGTR). The proposed UFSAR changes included with this submittal are based on analyses and evaluations of the impact of increased letdown flow rate on the RCS iodine concentration and normal appearance rate. Other limiting accidents are unaffected because they do not model iodine spikes (Waste Gas Decay Tank Failure or Fuel Handling Accident) or the iodine spike source term is insignificant compared to the fuel damage source term (LOCA, Control Rod Ejection, Locked Rotor).

Beaver Valley Event Notification #35359 identified potentially non-conservative assumptions with respect to letdown flow used in calculating the iodine appearance rate modeled in post-accident radiation dose analyses. Westinghouse issued a Nuclear Safety Advisory Letter, "NSAL-00-04: Nonconservatisms in Iodine Spiking Calculations," discussing this issue and several other potential issues:

- Letdown flow rate - the original analyses were based on a letdown flow rate of 60 gpm. Plant Farley currently operates with a maximum normal flow rate of approximately 130 gpm. For any assumed initial RCS dose equivalent iodine (DEI) concentration, a higher letdown flow rate would be balanced by a higher normal appearance rate to maintain the assumed equilibrium DEI concentration.
- Letdown demineralizer efficiency - a higher demineralizer efficiency removes more iodine which would be balanced by a higher normal appearance rate to maintain the assumed equilibrium DEI concentration. This is not applicable to Plant Farley as the original analyses assumed 100% removal.
- Primary coolant leakage - higher RCS leakage also removes more iodine which would be balanced by a higher normal appearance rate to maintain the assumed equilibrium DEI concentration. This is not applicable to Plant Farley as a value bounding the Technical Specifications limits for unidentified, identified, and primary to secondary leakage (11.3 gpm) was assumed in the original analyses.
- Letdown flow rate uncertainty - uncertainty in the flow rate effectively increases the potential removal which would be balanced by a higher normal appearance rate to maintain the assumed equilibrium DEI concentration. Plant Farley currently operates with a maximum normal flow rate of approximately 130 gpm. The measured flow plus indicated uncertainty for Plant Farley was calculated to increase the letdown flow to a maximum value of 145 gpm for any measurement/indication loop.
- Primary coolant mass - a larger primary coolant mass increases the initial inventory, which increases the normal appearance rate. This is not applicable to Plant Farley as a conservatively large RCS mass was assumed in the original analyses.

The accident initiated iodine spike modeled for the MSLB and SGTR increases relative to the increase in letdown flow rate (from 60 gpm plus 11.3 gpm to 145 gpm plus 11.3 gpm), affecting the radiological analyses in Sections 15.4.2 and 15.4.3 of the UFSAR. In addition, the higher

letdown flow rate may impact the RCS activity assumed for radwaste system design and shielding design discussed in Chapters 11 and 12 of the UFSAR.

For the RCS activity model discussed in UFSAR Section 11.1, an increase in the letdown flow rate will result in a decrease in the RCS activity concentrations. The GALE computer code analysis was re-performed for the 145 gpm letdown flow rate and the normal operations RCS activities. The resultant liquid and gaseous releases were confirmed to decrease. Therefore, the current descriptions in UFSAR Chapter 11 remain bounding. The attached UFSAR markups for Chapter 11 reflect the results of the re-analysis based on replacement steam generators. The higher primary and secondary mass for replacement steam generators is conservative. The containment volume was increased slightly for conservatism.

The source terms for the cumulative radiation dose for 40 years of normal operations described in Section 3.11 of the UFSAR and for shielding design described in UFSAR Chapter 12 are based on 1% defective fuel. These liquid and gaseous process activities would also decrease with an increasing letdown flow rate. Technical Specifications for the RCS limit the DEI activity to a maximum of 0.3 μCi/gm, which is less than the source term recommended for shielding in Regulatory Guide 8.8 and much smaller than the original value used for shielding (approximately 3.2 μCi/gm based on the activities shown in UFSAR Table 11.1-2). Although the process flow rate may increase by approximately 145/60, the input activity (concentration times flow rate) for components which concentrate activity, e.g. filters, demineralizers, and evaporators, originally used to calculate shielding source terms remains bounding. The letdown flow rate does not impact core activity so the post-LOCA source terms are not affected. Therefore, the radiation protection features and programs described in UFSAR Section 3.11 and Chapter 12 remain acceptable.

The accident initiated iodine spike for the MSLB and SGTR is modeled as 500 times the normal appearance rate. Since the normal appearance rate is in equilibrium with the normal cleanup rate, the increase in letdown flow will result in an increase in the normal appearance rate and hence in the accident initiated iodine spike. This increased iodine spike impacts the radiological doses calculated for the original steam generators. The iodine spike was recalculated and the results are shown with the analysis input values in the attached Tables 1 and 2. Note that UFSAR Table 15.4-23 erroneously indicated 8 hours to isolate the faulted steam generator. This value has been corrected to 30 minutes in the attached markups for consistency with the SGTR analysis.

For the original steam generators, which employ alternate repair criteria (ARC) to demonstrate acceptable steam generator tube performance for an MSLB, the allowable ruptured steam generator leak rate was reduced from the currently approved value of 11.8 gpm[1] to 5 gpm, and the RCS DEI^{131} was reduced from the previously analyzed value of 0.5 μCi/gm to the current Technical Specifications value of 0.3 μCi/gm. Using the revised iodine spike source terms with this lower allowable leakage and initial source term, the affected offsite doses were recalculated. Note that because of the frequency with which full UFSAR revisions are published, the attached UFSAR markups for Chapter 15 reflect a change from the previously approved ARC leakage of 23.8 gpm[2] instead of a change from the currently approved value of 11.8 gpm.

The resultant offsite thyroid doses are shown in Table 3 and compared to the previously submitted results. Control room doses for a LOCA were previously shown to be limiting[3]. The small changes in offsite thyroid doses for MSLB and SGTR would have a similarly small impact on control room doses. The limiting control room thyroid doses for a LOCA remain unchanged and continue to meet GDC 19 limits. The iodine spike has an insignificant effect on the whole body or skin doses. All doses meet the applicable acceptance criteria from the Standard Review Plan, NUREG-0800.

(1) NRC letter approving an I-131 limit of 0.3 μCi/gm, dated May 10, 1999, "Issuance Of Amendments - Joseph M. Farley Nuclear Plant, Units 1 and 2 (TAC Nos. MA5119 and MA5120)."

(2) NRC letter approving ARC leakage limit of 23.8 gpm, dated October 29, 1997, "Issuance Of Amendments - Joseph M. Farley Nuclear Plant, Units 1 and 2 (TAC Nos. MA99610 and MA99612)."

(3) NRC letter approving Power Uprate, dated April 29, 1998, "Issuance of Amendments - Joseph M. Farley Nuclear Plant, Units 1 and 2 (TAC Nos. M98120 and M98121)," Table 3.5.2-3.

Table 1 - Original Steam Generators MSLB Input Parameters

Core thermal power (MWt)	2,831
RCS volume (ft^3)	1.023×10^4
Steam generator tube leak rate prior to accident and intact generator leak rate for 8 h following accident	150 gpd / generator
Ruptured steam generator leak rate for 30 minutes following accident	5 gpm
Offsite power	Lost
Fuel defects (%)	1 for noble gases[a]
Iodine partition factor for steam release from faulted steam generator	1.0
Iodine partition factor in intact steam generators prior to and during accident	0.1
Time to isolate faulted steam generator (min)	30
Duration of plant cooldown by secondary system after accident (h)	8
Steam release from faulted steam generator (lb)(min)	473,000 (0-30)
Steam release from two intact steam generators (lb)(h)	339,000 (0-2) 730,000 (2-8)
Feedwater flow to two intact steam generators (lb)(h)	442,000 (0-2) 791,000 (2-8)
Meteorology	See UFSAR Appendix 15B

a. A pre-existing iodine spike of 18 μCi/gm DEI^{131} or an accident initiated iodine spike 500 times the normal appearance rate is assumed. The accident initiated iodine spike is based on an initial RCS DEI^{131} concentration of 0.3 μCi/gm, 145 gpm letdown flow rate, and bounding RCS leakage of 11.3 gpm. This yields an iodine spike source term of

I^{131}	2,938 Ci/hr
I^{132}	3,531 Ci/hr
I^{133}	5,795 Ci/hr
I^{134}	5,180 Ci/hr
I^{135}	4,627 Ci/hr.

Table 2 - Original Steam Generator SGTR Input Parameters

Core thermal power (MWt)	2,831
RCS volume (ft^3)	1.023×10^4
Steam generator tube leak rate prior to accident and intact generator leak rate 8 h following accident	150 gpd / generator
Offsite power	Lost
Fuel defects (%)	1 for noble gases[a]
Iodine partition factors for secondary side water in steam generators during accident	0.1
Time to isolate defective steam generator (min)	30
Duration of plant cooldown by secondary system after accident (h)	8
Steam release from ruptured steam generator	73,300 lb
Steam release from two intact steam generators	400,000 lb (0-2 hr) 889,000 lb (2-8hr)
Feedwater flow to two intact steam generators	320,000 lb (0-2 hr) 936,000 lb (2-8 hr)
Reactor coolant released to the ruptured steam generator	150,000 lb
Meteorology	See UFSAR Appendix 15B

a. A pre-existing iodine spike of 18 μCi/gm DEI^{131} or an accident initiated iodine spike 500 times the normal appearance rate is assumed. The accident initiated iodine spike is based on an initial RCS DEI^{131} concentration of 0.3 μCi/gm, 145 gpm letdown flow rate, and bounding RCS leakage of 11.3 gpm. This yields an iodine spike source term of

I^{131}	2,938 Ci/hr
I^{132}	3,531 Ci/hr
I^{133}	5,795 Ci/hr
I^{134}	5,180 Ci/hr
I^{135}	4,627 Ci/hr.

Table 3 - Accident Initiated Iodine Spike Offsite Thyroid Doses (REM)

	EAB Revised Letdown	EAB[1-2] Power Uprate	LPZ Revised Letdown	LPZ[1-2] Power Uprate
Original Steam Generators (Power Uprate)				
MSLB	17	17	26	26
SGTR	3.2	2.8	1.9	1.7

(1) Letter to NRC, dated 2/14/97, "Joseph M. Farley Nuclear Plant, Facility Operating Licenses and Technical Specifications Change Request for Power Uprating."

(2) Letter to NRC, dated 3/16/98, "Joseph M. Farley Nuclear Plant, Response to Request for Additional Information Related to Power Uprate, Facility Operating Licenses and Technical Specifications Change Request."

Attachment 2

Joseph M. Farley Nuclear Plant
Final Safety Analysis Report Change Request
for Original Steam Generators on Unit 2
Iodine Spiking due to Increased Letdown Flow Rate

Significant Hazards Evaluation

Joseph M. Farley Nuclear Plant
Final Safety Analysis Report Change Request
for Original Steam Generators on Unit 2
Iodine Spiking due to Increased Letdown Flow Rate

Significant Hazards Evaluation

Introduction and Background

Southern Nuclear Operating Company (SNC) has completed a comprehensive review of the Beaver Valley Event Notification #35359 and Westinghouse Nuclear Safety Advisory Letter, "NSAL-00-04: Nonconservatisms in Iodine Spiking Calculations," which identified potentially non-conservative assumptions with respect to letdown flow used in calculating the iodine appearance rate modeled in post-accident radiation dose analyses. In addition, letdown demineralizer efficiency, letdown flow rate uncertainty, primary coolant leakage, and primary coolant mass assumptions were reviewed.

The values of these parameters used in design basis analyses were reviewed and changes to the letdown flow rate and flow rate uncertainty are required to bound current operations. Changes to the previous values of letdown demineralizer efficiency, primary coolant leakage, and primary coolant mass are not required. The impacts of the changes on radiological consequences of accidents, offsite doses from normal operation and anticipated operational occurrences, safety-related electrical equipment qualification doses, and plant shielding were examined.

Proposed Changes

The original FSAR described analyses were based on a letdown flow rate of 60 gpm. Plant Farley currently operates with a maximum normal flow rate of approximately 130 gpm. Uncertainty in the flow rate effectively increases the potential removal. The measured plus indicated uncertainty for Plant Farley was calculated to increase the letdown flow to a maximum value of 145 gpm for any measurement/indication loop. For any assumed initial RCS dose equivalent iodine (DEI) concentration, a higher letdown flow rate would be balanced by a higher normal appearance rate to maintain the assumed equilibrium DEI concentration.

An accident initiated iodine spike equal to 500 times the normal appearance rate was used to re-evaluate the offsite doses for a steam generator tube rupture (SGTR) and main steam line break (MSLB). For the original steam generators, which employ alternate repair criteria to demonstrate acceptable steam generator tube performance for an MSLB, the allowable ruptured steam generator leak rate was reduced from the currently approved value of 11.8 gpm to 5 gpm, and the RCS DEI^{131} was reduced from the previously analyzed value of 0.5 μCi/gm to the current Technical Specifications value of 0.3 μCi/gm. Using the revised iodine spike source terms with this lower allowable leakage and initial source term, the affected offsite doses were recalculated.

For the RCS activity model discussed in UFSAR Section 11.1, an increase in the letdown flow rate will result in a decrease in the RCS activity concentrations. The GALE computer code analysis was re-performed for the 145 gpm letdown flow rate and the normal operations RCS activities. The resultant liquid and gaseous releases were confirmed to decrease. Therefore, the current descriptions in UFSAR Chapter 11 remain bounding.

The source terms for the cumulative radiation dose for 40 years of normal operations described in Section 3.11 of the UFSAR and for shielding design described in UFSAR Chapter 12 are based

on 1% defective fuel. These liquid and gaseous process activities would decrease with an increasing letdown flow rate. Technical Specifications for the RCS limit the DEI activity to a maximum of 0.3 μCi/gm, which is less than the source term recommended for shielding in Regulatory Guide 8.8 and much smaller than the original value used for shielding (approximately 3.2 μCi/gm based on the activities shown in UFSAR Table 11.1-2). Although the process flow rate may increase, the input activity (concentration times flow rate) for components which concentrate activity, e.g. filters, demineralizers, and evaporators, originally used to calculate shielding source terms remains bounding. The letdown flow rate does not impact core activity so the post-LOCA source terms are not affected.

10 CFR 50.92 Evaluation

The impacts of the proposed UFSAR changes in letdown flow rate on offsite doses from normal operation and anticipated operational occurrences, safety-related electrical equipment qualification doses, and plant shielding were evaluated. Normal operations releases calculated with the GALE computer code decreased, and the source terms used for plant shielding and doses to safety-related electrical equipment are unchanged or decreased. Therefore, it is concluded that no changes to the evaluations and analyses for these topics previously submitted in support of power uprate or replacement steam generators are required.

The change in letdown flow rate results in increased accident-induced iodine spike source terms. The MSLB and SGTR offsite dose consequences were reanalyzed for the increased iodine spike source terms and included a reduction of the initial RCS activity analyzed to the Technical Specifications limit and a coincident reduction in the allowable ruptured steam generator primary to secondary leakage to 5 gpm. The results of the offsite dose analyses show only slight changes from results previously submitted for power uprate or replacement steam generators, and all results continue to meet the acceptance criteria as described in the safety evaluation reports for power uprate and steam generator replacement. The limiting control room thyroid doses for a LOCA remain unchanged and continue to meet GDC 19 limits.

Based on the analyses and evaluations performed for the increase in letdown flow rate, the following conclusions can be reached with respect to 10 CFR 50.92.

1. The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated in the UFSAR. The comprehensive engineering review included evaluations or re-analysis of all accident analyses. Calculations for letdown flow measurement and indication have verified the acceptability of the analyzed letdown flow rate. The letdown flow rate does not initiate any accident; therefore, the probability of an accident has not been increased. All dose consequences have been analyzed or evaluated with respect to the proposed changes, and all acceptance criteria continue to be met. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident than any accident already evaluated in the UFSAR. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The changes have no adverse effects on any safety-related system and do not challenge the performance or integrity of any safety-related system. Therefore, all accident analyses criteria continue to be met and these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety. All analyses and evaluations using letdown flow rate as an input have been revised to reflect the proposed value. The calculations are based on FNP instrumentation and test methods and include uncertainty allowances. The evaluations and analyses results demonstrate applicable acceptance criteria are met. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the previous information and on the analyses and evaluations performed, the proposed changes do not involve a significant hazards consideration as defined in 10 CFR 50.92.

Attachment 3

Joseph M. Farley Nuclear Plant
Final Safety Analysis Report Change Request
for Original Steam Generators on Unit 2
Iodine Spiking due to Increased Letdown Flow Rate

Markups of Affected UFSAR Pages

TABLE 11.1-1 (SHEET 1 OF 3)

PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES[c]

Parameter	Value
Reactor power (MWt)	2774
Clad defects, as a percent of rated core thermal power being generated by rods with clad defects	1.0 (d)
Reactor coolant liquid volume (ft^3)	9723.3
Reactor coolant full power average temperature (°F)	577
Normal purification flowrate (gal/min)	60 (d)
Effective cation demineralizer flow (gal/min)	6.0 (d)
Volume tank volumes	
Vapor (ft^3)	175
Liquid (ft^3)	125
Fission product escape rate coefficients	
Noble gas isotopes (s^{-1})	6.5×10^{-8}
Br, I, and Cs isotopes (s^{-1})	1.3×10^{-8}
Te isotopes (s^{-1})	1.0×10^{-9}
Mo isotopes (s^{-1})	2.0×10^{-9}
Sr and Ba isotopes (s^{-1})	1.0×10^{-11}
Y, La, Ce, and Pr isotopes (s^{-1})	1.6×10^{-12}
Mixed bed demineralizer decontamination factors	
Noble gases and Cs-134, Cs-136, Cs-137, Y-90, Y-91, and Mo-99	1.0
All other isotopes including corrosion products	10.0

TABLE 11.1-1 (SHEET 3 OF 3)

Isotope	Partition Coefficient[b]
I-131	100
I-132	100
I-133	100
I-134	100
I-135	100

d. Evaluations of the impact of increasing letdown flow to 145 gpm (and cation demineralizer flow to 14.5 gpm) indicate this original combination of parameters continues to give conservatively high design basis source terms.

a. Fraction of inlet isotope concentration stripped off in the volume control tank.

b. Ratio between isotopic concentration in the liquid phase and isotope concentration in the vapor phase.

c. Reviewed for power uprate, RCS activities in table 11.1-2 remain bounding for power uprate

TABLE 11.1-7 (SHEET 1 OF 3)

PARAMETERS SPECIFIED BY REGULATORY GUIDE 1.112 APPENDIX B (INPUT PARAMETERS FOR THE GALE COMPUTER CODE)[3]

Description	Value
Thermal power level (MWt)	2785
Mass of primary coolant (lb)	4.2 ~~4.0×10^5~~
Primary system letdown rate (gal/min)	60[a]
Letdown cation demineralizer flowrate (gal/min)	6.0[a]
Number of steam generators	3
Total steam flow (lb/h)	12.3×10^6
Mass of steam in each steam generator (lb)	6.5 ~~6.3×10^5~~
Mass of liquid in each steam generator (lb)	1.0 ~~9.1×10^5~~
Total mass of secondary coolant	1.52×10^6
Total blowdown rate (lb/h)	3.75×10^4
Condensate demineralizer regeneration time	0.0
Condensate demineralizer flow fraction	0.0
Maximum radwaste dilution flow (gal/min)	16.0×10^3
Shim Bleed	
Shim bleed flowrate (gal/day)	1.56×10^3
Decontamination factor for I	10^5
Decontamination factor for Cs and Rb	4×10^4
Decontamination factor for others	10^6
Collection time (day)	1.03
Process and discharge time (day)	0.148
Fraction discharged	1.0
Equipment Drains	
Equipment drains flowrate (gal/day)	250
Fraction of reactor coolant activity	0.005
Decontamination factor for I	10^3
Decontamination factor for Cs and Rb	10^4
Decontamination factor for others	10^4
Collection time (day)	1.03
Process and discharge time (day)	0.148
Fraction discharged	1.0
Clean Waste	
Clean waste input flowrate (gal/day)	1.64×10^2
Fraction of reactor coolant activity	0.005
Decontamination factor for I	10^3
Decontamination factor for Cs and Rb	10^4
Decontamination factor for others	10^4

TABLE 11.1-7 (SHEET 3 OF 3)

Description	Value
Gaseous Waste System	
Holdup time for xenon (day)	90
Holdup time for krypton (day)	90
Fill time of decay tanks for gas stripper	0.0
Gas waste system: HEPA?	Yes
Auxiliary building: charcoal?	Yes
Auxiliary building: HEPA?	Yes
Containment volume (ft^3)	2.15 ~~2.03~~ x 10^6
Containment atmosphere cleanup rate (ft^3/min)	20 x 10^3
Containment shutdown purge: charcoal?, HEPA?	Yes, Yes
Number purge per year	8
Containment normal purge rate (ft^3/min); charcoal?, HEPA?	0; yes, yes
Fraction of iodine released from blowdown tank vent	0.05
Fraction of iodine released from main condenser air ejector	0.1
Detergent waste decontamination factor	0.1

a. Evaluation of the impact of increasing letdown flow to 145 gpm and cation demineralizer flow to 14.5 gpm indicate the releases shown in Tables 11.2-7, 11.2-8, 11.3-9 and 11.3-10 and the resultant doses shown in Tables 11.2-9 and 11.3-11 remain bounding.

L. The primary pressure remains constant at 2235 psig for 0 to 2 h and decreases linearly to atmospheric from 2235 psig during the period of 2 to 8 h.

M. The 0 to 2 and 2- to 8-h atmospheric diffusion factors given in appendix 15B and the 0- to 8-h breathing rate of 3.47×10^{-4} m^3/s are applicable.

The steam releases to the atmosphere for the steam line break are given in table 15.4-23.

LIMITING

The gamma, beta, and thyroid doses for the steam line break accident for the conservative analysis at the site boundary and the low-population zone are given in table 15.4-23. The doses from this accident are within the NRC acceptance criteria described in NUREG 0800, section 15.1.5.

The potential for uncovery of the tubes in the intact steam generators during the event has also been evaluated for impact on doses. The tube uncovery was assumed to exist for the first 1/2 h of the accident, and the tube leakage locations were assumed to all be near the top of the tube bundle and, thus, subject to the uncovery. With the primary to secondary leakage entering the vapor space, no credit was taken for mixing with the secondary coolant, nor was credit taken for a partition factor within the steam generator (i.e., the primary coolant was assumed to be released directly to the environment). The uncovery does not impact the release of noble gases to the environment; thus, the gamma and beta doses are not affected. The uncovery does result in an increase in the accident releases of iodine. The effect on the conservative thyroid dose at the site boundary (assuming a primary to secondary leak rate of 150 gpd per steam generator is that this dose remains well within the limits as defined in 10 CFR 100.

15.4.2.1.5 Environmental Consequences of a Postulated Steam Line Break Associated with the Use of Steam Generator Alternate Repair Criteria (ARC) for Outside Diameter Stress Corrosion Cracking (ODSCC)

The use of the voltage-based repair criteria for repair of steam generator tubes affected by ODSCC has been approved in accordance with Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking." In order to predict the primary-to-secondary leakage in the event of a main steam line break, a model

for calculating the leakage from the faulted steam generator during a postulated main steam line break has been developed. This model has two major components: (1) a model predicting the probability that a given indication will leak as a function of voltage; and (2) a model predicting leak rate as a function of indication voltage, given that leakage occurs. The predicted leakage at the end-of-cycle is based on a predicted steam generator indication voltage distribution determined by consideration of eddy current uncertainty and growth rates of indications over the operational cycle.

The results of the calculation of predicted leakage is compared against a leakage limit intended to ensure that the NRC acceptance criteria listed below are met:

A. doses for the Exclusion Area Boundary (EAB) and Low-Population Zone (LPZ) are within the acceptance criteria of Standard Review Plan 15.1.5, Appendix A of NUREG-0800; and

B. the control room operator thyroid dose would be less than the acceptance criteria of General Design Criteria 19.

The leakage limit for primary-to-secondary leakage from the faulted steam generator in the event of a main steam line break is ~~23.8~~ gpm. The assumptions used to determine this limit are described in table 15.4-23.

15.4.2.2 Major Rupture of a Main Feedwater Pipe

15.4.2.2.1 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater pipe large enough to prevent the addition of

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(NOT TO BE INCORPORATED)
NOTE: INTERIM REDUCTION TO 11.8 gpm NOT REFLECTED IN CURRENT PUBLISHED FSAR REVISION.

TABLE 15.4-23 (SHEET 1 OF 2)

PARAMETERS USED IN STEAM LINE BREAK ANALYSES

	Conservative Analysis
Core thermal power (MWt)	2831
Steam generator tube leak rate prior to accident and initial 8 h following accident (gal/min)	150 gpd per generator
Offsite power	Lost
Fuel defects (%)	1[a]
Iodine partition factor for initial steam release from defective steam generator	1.0
Iodine partition factor in nondefective steam generators prior to and during accident	0.1
Time to isolate defective steam generator (h) (MIN)	~~0~~ 30
Initial steam release from defective steam generator (lb)(min)	473,000 (0-30)
Steam release from two nondefective steam generators (lb)(h)	339,000 (0-2) 730,000 (2-8)
Feedwater flow to two nondefective steam generators (lb)(h)	442,000 (0-2) 791,000 (2-8)
Meteorology	Accident (see Appendix 15B)

TABLE 15.4-23 (SHEET 2 OF 2)

Off-Site Dose (Rem)	Pre-Accident Iodine Spike	Accident-Initiated Iodine Spike
Site Boundary		
Thyroid	~~7.0~~	~~0.9~~ 17
Whole body	~~9×10^{-3}~~	~~9×10^{-3}~~
B-skin	~~5×10^{-3}~~	~~5×10^{-3}~~
LPZ		
Thyroid	~~3.0~~	~~3.0~~ 26
Whole body	~~4×10^{-3}~~	~~4×10^{-3}~~
B-skin	~~2×10^{-3}~~	~~2×10^{-3}~~

NON-LIMITING

Assumptions Used for the NRC Analysis to Determine Leakage Limit Associated with Use of Voltage-Based ARC for ODSCC Cracking in Steam Generator Tubes:

A.	Transient Primary Coolant Concentration of ~~9~~ 18 µCi/gm of dose equivalent I-131.	
B.	Limits For Equilibrium Dose Equivalent I-131 in the Primary and Secondary Coolant	
	Primary Coolant Dose Equivalent I-131 Concentration	~~0.15~~ 0.3 µCi/gm
	Secondary Coolant Dose Equivalent I-131 Concentration	0.1 µCi/gm
C.	Primary-to-Secondary Leak Rate Limits	
	Primary-to-Secondary Leak Rate, Maximum for Any Steam Generator	150 gpd
	Primary-to-Secondary Leak Rate, Total for All Steam Generators	450 gpd
D.	Maximum Primary-to-Secondary Leak Rate in the Faulted and Intact Steam Generators	
	Faulted Steam Generator	~~23.6~~ 5 gpm
	Intact Steam Generators	0.1 gpm/steam generator
E.	Iodine Partition Factor	
	Faulted Steam Generator	1
	Intact Steam Generator	0.1
	Primary-to-Secondary Leakage	1.0
F.	Steam Released to the Environment	(See Sheet 1)
H.	RCS LETDOWN FLOW RATE	145 gpm
G.	Atmospheric Dispersion Factors	
	EAB (0-2 hours)	6.5×10^{-4}

[SEE NOTE ON FSAR PAGE 15.4-43a (NOT TO BE INCORPORATED)]

a. A pre-existing iodine spike of ~~30~~ 18 µCi/gm or an accident initiated iodine spike 500 times the normal appearance rate is assumed.

TABLE 15.4-24 (SHEET 1 OF 2)

PARAMETERS USED IN STEAM GENERATOR TUBE RUPTURE ANALYSES

	Conservative Analysis
Core thermal power (MWt)	2831
Steam generator tube leak rate prior to and during accident (gpd/generator)	150
Offsite power	Lost
Fuel defects (%)	1[a]
Iodine partition factors in nondefective steam generators prior to and during accident	0.1
Iodine partition factor in defective steam generator prior to and during accident	0.1
Time to isolate defective steam generator (min)	30
Duration of plant cooldown by secondary system after accident (h)	8
Steam release from defective steam generator (lb)(min)	73,300 (0-30)
Steam release from two nondefective steam generators (lb)(h)	400,000 (0-2) 889,000 (2-8)
Feedwater flow to two nondefective steam generators (lb)(h)	320,000 (0-2) 936,000 (2-8)
Reactor coolant released to the defective steam generator (lb)	150,000

No CHANGES - INFORMATION ONLY

TABLE 15.4-24 (SHEET 2 OF 2)

Meteorology	Conservative Analysis Accident (see appendix 15B) Pre-Accident Iodine Spike	Accident Initiated Iodine Spike
Off-Site Dose (Rem)		
Site Boundary		
Thyroid	19.0	~~2.8~~ 3.2
Whole body	0.2	0.1
B-skin	0.2	0.2
LPZ		
Thyroid	7.7	~~1.7~~ 1.9
Whole body	0.1	0.1
B-skin	0.2	0.1

a. Pre-accident iodine spike 60 times the Technical Specification limit or accident initiated iodine spike 500 times the normal appearance rate.

BASED ON AN INITIAL RCS DEI-131 CONCENTRATION OF 0.3 μCi/gm AT A LETDOWN FLOW OF 145 gpm.