September 30, 2004

Mr. L. M. Stinson
Vice President - Farley Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MC4186 AND MC4187)

Dear Mr. Stinson:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 166 to Facility Operating License No. NPF-2 and Amendment No. 158 to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated August 25, 2004, as supplemented by letter dated September 27, 2004.

The amendments address the control room habitability guidance of Regulatory Guide 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," by revising Limiting Condition for Operation 3.7.10, "Control Room Emergency Filtration/Pressurization System (CREFS)" and TS 5.5.11, "Ventilation Filter Testing Program. The amendments also add a new section, TS 5.5.18, "Control Room Integrity Program (CRIP)."

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

Sincerely,

/RA/

Sean E. Peters, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosures:
1. Amendment No. 166 to NPF-2
2. Amendment No. 158 to NPF-8
3. Safety Evaluation

cc w/encls: See next page

September 30, 2004

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cc w/encls: See next page
DISTRIBUTION: See next page
Package Number: ML
Amendment Number: ML042780424
Tech Spec Number: ML

OFFICE	PDII-1/PM	PDII-1/LA (A)	SPSB/SC	EMCB/SC	IROB/SC	OGC	PDII-1/SC (A)
NAME	SPeters	DClarke	RDennig	LLund	TBoyce	DReddick	MRoss-Lee
DATE	9/30/04	9/30/04	09/29/04	09/27/04	9/30/04	09/28/04	9/30/04

OFFICIAL RECORD COPY

Joseph M. Farley Nuclear Plant, Units 1 & 2

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SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MC4186 AND MC4187)

Date: September 30, 2004

PUBLIC
PDII-1 R/F
EHackett
MRoss-Lee
SPeters
DClarke
OGC
ACRS/ACNW
GHill (4)
TBoyce
WBeckner
RidsRgn2MailCenter
DLPMDPR
RDennig
MBlumberg
JHayes
CHarbuck
RHarvey
HWalker
YDiaz-Castillo

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 166
License No. NPF-2

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated August 25, 2004, as supplemented by letter dated September 27, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 166, are hereby incorporated in the license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Mary Jane Ross-Lee, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 30, 2004

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 158
License No. NPF-8

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated August 25, 2004, as supplemented by letter dated September 27, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-8 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 158, are hereby incorporated in the license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Mary Jane Ross-Lee, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 30, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 166

TO FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

ATTACHMENT TO LICENSE AMENDMENT NO. 158

TO FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

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

Remove	Insert
3.7.10-1	3.7.10-1
3.7.10-2	3.7.10-2
---	3.7.10-3
---	3.7.10-4
5.5-8	5.5-8
5.5-9	5.5-9
5.5-14	5.5-14
---	5.5-15
B 3.7.10-1	B 3.7.10-1
B 3.7.10-2	B 3.7.10-2
B 3.7.10-3	B 3.7.10-3
B 3.7.10-4	B 3.7.10-4
B 3.7.10-5	B 3.7.10-5
B 3.7.10-6	B 3.7.10-6
B 3.7.10-7	B 3.7.10-7
B3.7.10-8	B.3.7.10-8

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 158 TO FACILITY OPERATING LICENSE NO. NPF-8

SOUTHERN NUCLEAR OPERATING COMPANY, INC., ET AL.

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC, Commission) dated August 25, 2004, as supplemented by letter dated September 27, 2004 (Refs. 1 and 2), the Southern Nuclear Operating Company, Inc. (SNC, the licensee) et al., submitted a request for changes to the Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2, Technical Specifications (TSs). The proposed changes would address the control room habitability guidance of Regulatory Guide (RG) 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," by revising Limiting Condition for Operation (LCO) 3.7.10, "Control Room Emergency Filtration/Pressurization System (CREFS)," and TS 5.5.11, "Ventilation Filter Testing Program." The changes would also add a new section, TS 5.5.18, "Control Room Integrity Program (CRIP)." The September 27, 2004, letter provided clarifying information that did not change the scope of the amendment request as originally noticed, and did not change the NRC staff's initial proposed no significant hazards consideration as published in the *Federal Register*.

SNC's request is related to a license amendment request dated August 29, 2003 (Ref. 3), in which the licensee proposed the use of the accident analysis assumptions of RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," in order to allow the containment equipment hatch to be open during core alterations and/or during movement of irradiated fuel assemblies. RG 1.195 states, "The guidance contained in this regulatory guide will supersede corresponding radiological analysis assumptions provided in other regulatory guides when used in conjunction with guidance that is in Regulatory Guide 1.196..." These amendments for FNP, Units 1 and 2, mark the first time a licensee incorporated the guidance of RG 1.195 and RG 1.196 and are being approved in parallel.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criterion (GDC) 19, "Control Room," requires that licensees maintain the control room in a safe condition under accident conditions. Under these conditions, the licensee must provide adequate radiation protection to permit access and occupancy of the control room. Section 100.11 of 10 CFR, "Determination of exclusion area, low population zone [LPZ] and population

center distance," on the other hand, establishes the dose limits for the exclusion area and for the public.

In order to show that the radiation doses to people onsite and offsite will meet the above regulatory requirements, licensees perform evaluations of accident radiation doses. Regulatory guidance for these evaluations is provided in the form of regulatory guides and standard review plans (SRP). The regulatory requirements on which the NRC staff based its review are contained in 10 CFR Part 50, Appendix A, GDC 19, as supplemented by the SRP, Section 6.4, "Control Room Habitability System," and 10 CFR 100.11. Except where the licensee proposed a suitable alternative, the NRC staff used the regulatory guidance provided in the following documents in performing this review.

- SRP Section 6.4
- SRP Section 6.5.2, "Containment Spray as a Fission Product Cleanup System"
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants"
- RG 1.195[1]
- RG 1.196
- RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactor"
- RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident [LOCA] for Pressurized Water Reactors"
- RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants"

The NRC staff also considered relevant information in the FNP Updated Final Safety Analysis Report (UFSAR), TSs, responses to Generic Letter (GL) 2003-01, "Control Room Habitability," and the August 29, 2003, amendment request (Ref. 3), as supplemented.

3.0 TECHNICAL EVALUATION

3.1 Radiological Analysis

SNC performed a re-analyses of each applicable radiological accident listed in RG 1.195 and determined using RG 1.196 (Regulatory Position 2.3.2) that the LOCA provided the limiting accident doses. Such re-analyses were required because of SNC's proposal to: 1) increase

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[1]RG 1.195 was used for reference only to provide the list of applicable accidents and for information contained in Section B, "Discussion." This safety evaluation (SE) references a FNP fuel-handling accident (FHA) analysis that used RG 1.195 for its methods and assumptions. The NRC staff did not use RG 1.195 in its review of any other FNP design-basis accidents (DBAs).

the amount of assumed unfiltered inleakage into the control room, 2) change the containment spray removal credit, and 3) change the atmospheric dispersion factors used. These changes would alter the releases from the postulated accidents and would alter the offsite and control room doses.

3.1.1 LOCA

SNC calculated the potential consequences of a postulated LOCA to the control room operators and to individuals located offsite at the exclusion area boundary (EBA) and LPZ. SNC postulated that the pathways for releases in the event of a LOCA were containment leakage and emergency core cooling system recirculation loop leakage discharged via the penetration room filter system. The licensee also assumed that at the time the LOCA occurred, the containment was being purged. The purge was assumed to be isolated within 6 seconds following the accident. The assumptions used by the NRC staff are presented in Tables 1 through 4. Specific issues regarding the LOCA analysis are discussed below.

3.1.1.1 Containment Sump pH

Following a LOCA, maintaining the containment sump water in an alkaline condition is needed to prevent dissolved radioactive iodine from being released to the containment atmosphere during the recirculation phase of the containment spray injection. Most of the iodine leaves the damaged core in an ionic form which is readily dissolved in the sump water. However, in an acidic environment some of the iodine becomes converted into elemental form which is much less soluble, causing re-evolution of iodine to the containment atmosphere. In order to prevent release of elemental iodine to the containment atmosphere after a LOCA, the sump pH has to be maintained equal to or higher than 7.

Therefore, in the LOCA analysis there is a need for determining containment sump water pH, which depends on the chemical species dissolved in the water. After a LOCA, certain chemical species could be released from the damaged core or generated in the radiation fields existing in the containment. The majority of the species introduced into the containment sump water are either acidic or basic and the resultant containment sump pH would depend on their relative concentrations and on the buffering action of the trisodium phosphate (TSP) added to the containment sump water.

The licensee developed a model to predict the change in pH of a system containing boric acid and certain chemical species when TSP is added. Although the licensee used a spreadsheet model to perform the calculations, enough information was provided to allow the NRC staff to perform its own independent verification of the results.

The licensee determined minimum and maximum values of pH for various temperature ranges. The results indicate a minimum pH of 7.72 and a maximum pH of 8.6 would be obtained as a result of a large break LOCA. Since the TSP provides a strong buffering action, the pH should remain above 7 for the duration of the 30-day transient. The NRC staff's independent verification demonstrated that the containment sump water pH would remain above 7, which is consistent with the licensee's submittal. Therefore, the NRC staff finds the licensee's analysis to be acceptable.

3.1.1.2 Iodine Spray Removal Coefficients

Radioactive iodine is released from the containment sump in three different forms: elemental, particulate and organic. There is no effective mechanism for removing organic iodide from the containment atmosphere. According to Section 6.5.2 of the SRP, it is conservative to assume that organic iodides are not removed by either spray or wall deposition. The licensee did not take credit for the removal of organic iodide. For elemental iodine, on the other hand, the licensee considered two distinct mechanisms by which radioactive iodine could be removed: containment sprays and natural deposition on containment walls. Particulate iodide is only removed by the containment spray. Removal of iodine is controlled by two types of parameters: those controlling rates of removal, called lambdas (λ), and those determining the maximum amount that can be removed, called decontamination factors (DF).

The licensee calculated spray removal coefficients for elemental and particulate iodine during the injection and recirculation phase. The licensee used the methodology described in WCAP-11611, "Methodology for Examination of the Containment Spray Additive," to calculate the spray removal coefficient for elemental iodine. The licensee's value for the spray removal coefficient for elemental iodine was $\lambda_s = 13.7\ hr^{-1}$, which is below the maximum acceptable value of $20\ hr^{-1}$ specified in the SRP. SNC introduced additional conservatism into the calculation of dose rates by using the value of $\lambda_s = 10\ hr^{-1}$ for both the injection and recirculation phases, which is lower than the value obtained by the calculation and lower than the maximum acceptable limit specified in the SRP. For the removal of elemental iodine by natural deposition, SNC used an analytical expression from Section 6.5.2 of the SRP. The licensee's value for the removal coefficient by natural deposition was $\lambda_w = 2.7\ hr^{-1}$.

For the spray removal coefficient for particulate iodide, the licensee used the analytical expression for spray removal as specified in Section 6.5.2 of the SRP. SNC's values for the spray removal coefficient for particulate iodide were $\lambda_p = 5.449\ hr^{-1}$ for the injection phase and $\lambda_p = 5.032\ hr^{-1}$ for the recirculation phase. The difference between λs for injection and recirculation phases was due to the spray flow for the injection phase being higher than that for the recirculation phase. The NRC staff performed hand calculations of the λ values for elemental, particulate and natural deposition spray removal coefficients using the equations set forth in Section 6.5.2 of the SRP. The NRC staff agrees with the licensee's results, and finds that the results are conservative because the values proposed by the licensee are lower than those obtained by the SRP calculation and are lower than the maximum limits specified in the SRP.

The DF for iodine in the containment is a function of the partition coefficient for iodine, the containment sump and sump overflow volume, and the containment building net free volume. The partition coefficient for iodine is a function of the sump water temperature, pH, and iodine content. According to the SRP, the maximum DF is 200 for elemental iodine. The licensee's value for the elemental iodine DF is 21. Because the removal mechanisms for organic and particulate iodides are significantly different and slower than that for elemental iodine, there is no need to limit the DF for organic and particulate iodides. The NRC staff performed hand calculations of the DF value for the elemental spray removal coefficient using the equation set forth in Section 6.5.2 of the SRP. The NRC staff's calculations were consistent with the values calculated by the licensee.

The NRC staff performed verification of the iodine removal coefficients and the DF by the licensee for removal of elemental and particulate iodine from the post-accident containment atmosphere. Based on its results, the NRC staff finds that all the values reported by the licensee in its submittal to be conservative and, therefore, acceptable.

3.1.1.3 Atmospheric Dispersion Estimates

3.1.1.3.1 Meteorological Data

To support the analysis of the LOCA dose consequences, SNC used 4 years of onsite hourly meteorological data collected during calendar years 2000 through 2003 to generate new control room atmospheric dispersion factors (χ/Q values). The resulting χ/Q values represent a change from those χ/Q values used in the current UFSAR analyses. The only new atmospheric dispersion factors generated for this amendment request were for the control room; existing UFSAR χ/Q values were used to evaluate doses for the EAB and LPZ.

SNC provided the 2000 through 2003 onsite hourly meteorological data in the form of hourly meteorological data files. The licensee stated that these data were collected by a meteorological monitoring program implemented in accordance with RG 1.23, “Onsite Meteorological Programs.” The data recovery rate for this period exceeded 90 percent. All releases were considered to be ground level releases. Wind data measured at 10.7 meters (35 feet) and 45.7 meters (150 feet) above ground-level were provided as input to the control room χ/Q analysis. Stability class was based on delta-temperature measurements made between the 61.0 meters (200 feet) and 10.7 meters (35 feet) levels on the onsite meteorological tower.

The NRC staff performed a quality review of the 2000 through 2003 onsite hourly meteorological data using the methodology described in NUREG-0917, “Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data.” The NRC staff also performed further confirmatory analysis. Examination of the data revealed that stable and neutral atmospheric conditions were generally reported to occur at night and unstable and neutral conditions during the day, as expected. Wind speed, wind direction, and stability class frequency distributions for each measurement channel were reasonably similar from year to year. However, a comparison of the 2000 through 2003 wind direction frequency distribution with the 1971 through 1975 wind direction frequency distribution presented in the FNP UFSAR Table 2.3-8B shows an apparent lack of southerly winds in the 2000 through 2003 data set. The licensee has suggested that the cooling towers may be interrupting the local on-site flow from the southerly direction. These towers were not operational during the 1971 through 1975 time frame.

In summary, the NRC staff has reviewed the available information relative to the onsite meteorological measurements program and the meteorological database provided by the licensee. On the basis of this review, the NRC staff concludes that, with the adjustments discussed in the next section to account for the apparent lack of southerly winds, these data provide an acceptable basis for making estimates of atmospheric dispersion for DBA assessments for the purposes of this amendment request.

3.1.1.3.2 Control Room Atmospheric Dispersion Factors

The licensee calculated control room air intake χ/Q values using 2000 through 2003 onsite meteorological data and the ARCON96 atmospheric dispersion computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). SNC determined the control room χ/Q values for releases from the FNP Units 1 and 2 reactor buildings and the reactor vents to each of the two (Unit 1 and 2) control room emergency air intakes. The licensee used the resulting bounding (highest) χ/Q values for each averaging interval in the subsequent dose analyses.

The NRC staff qualitatively reviewed the inputs to the ARCON96 computer runs and found them generally consistent with site configuration drawings and NRC staff practice. Specific areas of note are as follows:

- The reactor building releases were modeled as ground-level area sources. The release heights were assumed to occur at the same height as the control room air intakes. Only the building area that was higher than the air intake height was considered in the wake analysis.

- The reactor vent releases were modeled as ground-level point sources. The differences in elevation between the release heights and the intake heights were taken into consideration. No building wake effects were considered.

The resulting control room emergency air intake χ/Q values were used to analyze emergency filtered pressurization air makeup through the control room emergency air intake as well as unfiltered inleakage into the control room.

To address NRC staff concerns regarding the apparent lack of southerly winds in the 2000 through 2003 onsite meteorological data set, the licensee constructed an additional meteorological data file by copying as-recorded December 1999 data from the SSE, S, and SSW directions into a 1-month data set. These data were repeated six times and added to the 2000 through 2003 data to generate a 4½-year data set with an overall frequency from these directions similar to that in the 1971 through 1975 data set. The licensee performed an ARCON96 modeling analysis using this 4½-year data set for all release-receptor combinations. The higher χ/Q values resulting from either the original 2000 through 2003 data set or the modified 4½-year data set were used in the subsequent dose analyses.

Note that the licensee also presented information concerning technical support center (TSC) χ/Q values associated with releases from the reactor buildings, reactor vents, and containment hatch doors in their response to the request for additional information letter dated September 27, 2004 (Ref. 2). These χ/Q values were not used to support this licensing action; instead, they were intended for use by the licensee in future amendment requests. Consequently, these TSC χ/Q values were not reviewed by the NRC staff as part of this amendment request.

In summary, the NRC staff reviewed the licensee's assessments of control room post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. The resulting control room χ/Q values are presented in Table 2. On the

basis of this review, the NRC staff concludes that these χ/Q values are acceptable for use in performing control room dose assessments for a LOCA.

3.1.1.3.3 Offsite Atmospheric Dispersion Factors

The licensee evaluated offsite doses using offsite (EAB and LPZ) χ/Q values presented in the FNP UFSAR Tables 2.3-12 and 15B-2. These values are presented in Tables 3 and 4 of this SE. They represent sector independent (overall site) five percentile χ/Q values derived from hourly records of onsite data from the period April 1971 through March 1972. Details on their calculation can be found in the FNP UFSAR Section 2.3.4.

The NRC staff has reviewed the licensee's use of existing UFSAR EAB and LPZ χ/Q values and has found them to be appropriate for the application in which they are being used. On the basis of this review, the NRC staff concludes that these χ/Q values are acceptable for use in the EAB and LPZ dose assessments.

3.1.1.4 Control Room Doses and Unfiltered Inleakage

The NRC staff is currently working toward resolution of generic issues related to control room habitability, in particular, the validity of control room inleakage rates assumed by licensees in analyses of control room habitability. The NRC staff issued GL 2003-01, "Control Room Habitability." SNC responded to this GL by letter dated August 25, 2004 (Ref. 4). In its response, SNC reported that inleakage testing using the ASTM E741 tracer gas methodology determined a control room unfiltered inleakage rate of 25 cfm during the pressurization mode (used for LOCA analysis), 33 cfm during the isolation mode, and 87 cfm during the normal alignment. For the LOCA analyses, SNC proposed using a value of 53 cfm, which includes an assumed value of 10 cfm for ingress and egress. The proposed value is conservative compared to the measured value (53 cfm - 10 cfm = 43 cfm. 43 cfm is greater than the 25 cfm measured).

Although the SNC response to the GL is still under review, the NRC staff has determined that there is reasonable assurance that the FNP control room will be habitable during DBAs, and that this amendment may be approved before the final resolution of the generic issue. The NRC staff based this determination on (1) the results of the tracer gas testing, and (2) the independent confirmatory calculations performed by the NRC staff. The acceptance of FNP's unfiltered inleakage assumption for the purpose of this amendment request does not establish that the NRC staff has found the August 25, 2004 response adequate. The NRC staff will respond to the licensee's GL response under separate cover once its review is complete.

3.1.1.5 Offsite Doses

The EAB, LPZ, and control room doses estimated by SNC for the LOCA were found to be acceptable. The NRC staff performed independent calculations and confirmed the FNP conclusions.

3.1.2 Analysis of Each Applicable Accident

The licensee stated that they comply with RG 1.196 including Regulatory Position 2.3.2 that provides NRC positions on determining the limiting doses to the control room operators. In

complying with RG 1.196, SNC factored all potential differences in the accidents and the different performance of the control room. Furthermore, the NRC staff performed an independent calculation for the FHA given the revised inleakage characteristics allowed by the proposed TS changes. The NRC staff's evaluation is provided by letter dated September 30, 2004 (Ref. 5). This evaluation shows that the FHA doses are less than the LOCA values. Lastly, the NRC staff reviewed the SE report for license amendments 137 and 129 for FNP, Units 1 and 2, respectively, dated April 29, 1998 (Ref. 6). In this SE report the NRC staff confirmed that the LOCA was the limiting accident for the control room operator dose. While some input parameters used in this analysis have changed since this conclusion was drawn (for example, greater unfiltered inleakage), the NRC staff considered the previous conclusions drawn in this document. Based upon the FNP stated compliance with Regulatory Position 2.3.2, the NRC staff's independent evaluation of the FHA, the previous analysis performed by the NRC staff for Amendments 137 and 129 cited above, and the relative change in the magnitude of the inleakage as measured by E741 testing since the Amendment 137 and 129 SEs, the NRC staff believes there is reasonable assurance that the LOCA remains the bounding accident for the control room. In this decision, the NRC staff relied heavily upon the FNP stated compliance with Regulatory Position 2.3.2 discussed above.

3.2 TS Changes

SNC proposed the following changes to TS 3.7.10:

1. Re-titling the specification as control room.

2. Specifying the LCO in terms of two CREFS trains and the control room envelope (CRE).

3. Revising the CONDITION of ACTION B.

4. Modifying the REQUIRED ACTIONS and COMPLETION TIMES of ACTION B.

5. Relocating the existing CONDITION of ACTION F.

6. Deleting the phrase "for reasons other than Condition B".

7. Modifying the REQUIRED ACTION and COMPLETION TIME for the relocated CONDITION of ACTION F.

8. Adding a CONDITION to ACTION E.

9. Modifying the operating time of CREFS in Surveillance Requirement (SR) 3.7.10.1.

10. Modifying the ΔP SR 3.7.10.4.

11. Adding SR 3.7.10.5 to verify CRE integrity.

In addition to the changes to TS 3.7.10, there were two other proposed changes which involved the TS Programs and Manuals Section. SNC proposed that a CRIP program be added as TS 5.5.18. SNC also proposed that the testing frequency of Ventilation Filter Testing Program (VFTP) of TS 5.5.11 be modified.

3.2.1 Background

TS 3.7.10 addresses the protection of the control room operators. This protection encompasses the control room environment and is commonly referred to as control room habitability. Control room habitability is a requirement arising from GDC 19 and from Three Mile Island (TMI) Action Item III.D.3.4 (NUREG-0737, Clarification of TMI Action Plan Requirements). Control room habitability is intended to ensure that a radiological event will not result in a dose to control room operators of 5 rem whole body or its equivalent. In addition, control room habitability is intended to ensure that neither a radiological nor a hazardous chemical nor a fire event will prevent the control room operators from controlling the reactor from either the control room or the alternate shutdown panel.

Control room habitability is typically provided by two components, the CRE and the CREFS. Each component provides protection which is unique and independent of each other. The CRE provides integrity. In this application, integrity is usually measured in terms of air inleakage rates into the CRE. The CREFS provides cleanup capability. It typically is composed of HEPA filters and charcoal adsorbers. Depending upon the design, they will either filter and adsorb one or both of the following: air being supplied to the CRE or air within the CRE.

Together, the CRE and CREFS provide the control room operators a habitable work environment. However, each is separate and unique in their function. The performance of the CREFS does not alter the condition of the CRE. In a similar manner, the condition of the CRE does not affect the performance of CREFS. If a breech has been made to the CRE and the breech is not associated with the ductwork or housing of the CREFS, the CREFS should remain capable of providing a certain flow rate and filter and adsorb at the licensing basis removal efficiency. If the CREFS is providing supply air to the CRE at a flow rate different than the licensing basis, it does not change the condition of the control room but it may change the inleakage characteristics of the CRE. However, the problem would not be the CRE but would be the CREFS.

In the past, TS 3.7.10 has focused on radiological challenges and, in particular, the CREFS. Therefore, the CREFS appeared to be the only cornerstone of control room habitability. However, as noted above, it is one of two cornerstones.

The proposed changes for FNP balance the specification so that now the specification is focusing on both the CREFS and the CRE. This balanced approach to control room habitability is appropriate. A majority of the TS changes are changes which address this balance. The following is a discussion of the proposed TS changes.

3.2.2 Re-titling the Specification

Currently, TS 3.7.10 is entitled, “Control Room Emergency Filtration/Pressurization System (CREFS)”. It was proposed that TS 3.7.10 be re-titled to “Control Room”. With the current TS title, there is the impression that control room habitability is solely dependent upon the performance of the CREFS. Some of this impression may arise from the existing CONDITION of ACTION B.

The CONDITION is described as two CREFS trains inoperable due to an inoperable CRE. The integrity of the CRE does not affect the operability of the CREFS. The capability of the CREFS

to provide a certain flow or to remove radioiodine and particulate is unaffected by the condition of the CRE. In a similar fashion, the condition of the CRE is not a function of the performance of the CREFS. An inoperable CRE most likely will have no effect on the capability of the CREFS to provide a given flow rate or to filter and adsorb with a given efficiency. A change in CREFS flow rate may change the leak rate from the CRE but it does not change the material condition of the CRE.

The re-titling of TS 3.7.10 to Control Room is appropriate. It more accurately reflects the focus of this TS, therefore, the NRC staff finds this change acceptable.

3.2.3 Specifying the LCO in terms of CREFS and the CRE

Currently the LCO specifies that two CREFS trains shall be OPERABLE. As noted above, it is recognized that control room habitability is provided by two CREFS trains and the CRE working in concert. Therefore, the LCO should require that two CREFS trains and the CRE be OPERABLE. The licensee proposed this requirement for FNP, and the NRC staff finds the proposal acceptable.

3.2.4 Revising the CONDITION of ACTION B

The existing CONDITION of ACTION B addresses two CREFS trains being inoperable while in MODE 1, 2, 3, and 4 due to an inoperable control room boundary. It has been proposed that the term CRE replace control room boundary.

As noted above, an inoperable CRE does not render the CREFS trains inoperable. The CRE and the two CREFS trains are two separate and distinct entities. With such a distinction, there needs to be an ACTION which addresses the CRE being inoperable in MODE 1, 2, 3 and 4 and while moving irradiated fuel assemblies and during CORE ALTERATIONS. The proposed change to ACTION B addresses this need. Therefore, the NRC staff finds this change acceptable.

3.2.5 Modifying the REQUIRED ACTIONS and COMPLETION TIMES of ACTION B

Commensurate with the change to the CONDITION of ACTION B is the necessity to change the REQUIRED ACTION and COMPLETION TIME. It was proposed that mitigating actions be initiated immediately when the CRE became inoperable and that there be 24 hours in which to restore the CRE to the OPERABLE state. This completion time is acceptable because of the small probability of a DBA occurring during the 24 hour period, even though the dose limits of GDC 19 could be exceeded despite the use of mitigating actions. If the CRE is not restored within 24 hours, it could remain inoperable for 30 days provided it was verified that the mitigating actions resulted in the facility still being able to meet GDC 19. The NRC staff finds the 30 day completion time acceptable because the actions taken on behalf of the facility are to ensure that GDC 19 continues to be met while providing sufficient time to correct most problems affecting CRE integrity.

3.2.6 Relocating the existing CONDITION of ACTION F

The existing ACTION F addresses two CREFS trains being inoperable while in MODE 1, 2, 3, and 4 for reasons other than an inoperable control room boundary. Since there no longer is a

link between the control room boundary (now described as CRE) being inoperable and the CREFS being inoperable, there needs to be an ACTION which addresses two CREFS trains being inoperable while in MODE 1, 2, 3, and 4. That CONDITION has been incorporated into ACTION C. The NRC staff finds this incorporation acceptable.

3.2.7 Deleting the phrase “for reasons other than Condition B”

As noted above, since there no longer is a link between the control room boundary being inoperable and the CREFS being inoperable, there is no longer a need for the phrase, “for reasons other than Condition B”. Therefore, the removal of this phrase from TS 3.7.10 is appropriate. The NRC staff finds this change acceptable.

3.2.8 Modifying the REQUIRED ACTION and COMPLETION TIME for the relocated CONDITION of ACTION F

Because of the merging of the original CONDITION F with CONDITION C, the associated REQUIRED ACTION must be updated. Under the original ACTION F, the REQUIRED ACTION was, “Enter LCO 3.0.3." LCO 3.0.3 requires the licensee to be in MODE 3 within 7 hours, MODE 4 within 14 hours, and MODE 5 within 37 hours. However, for consistency, the licensee proposed changing the REQUIRED ACTION to be consistent with the original CONDITION C REQUIRED ACTION of being in MODE 3 within 6 hours and MODE 5 within 36 hours. Because this change is conservative, the NRC staff finds it acceptable.

3.2.9 Adding a CONDITION to ACTION E

The existing ACTION E addresses the CONDITION where two CREFS trains are inoperable during the movement of irradiated fuel assemblies and during CORE ALTERATIONS. As noted above, with the addition of the requirement to have an OPERABLE CRE, there needs to be an ACTION which addresses the CONDITION of an inoperable CRE during the movement of irradiated fuel assemblies and during CORE ALTERATIONS. It was proposed to add such a CONDITION to the present ACTION E. The NRC staff finds this change acceptable.

3.2.10 Modifying the operating time of CREFS in SR 3.7.10.1

The licensee has proposed that the operating time of the CREFS for SR 3.7.10.1 be changed from 10 hours to 15 minutes. Such a change is consistent with the guidance in Revision 3 of RG 1.52, therefore, the NRC staff finds the proposed change acceptable.

3.2.11 Modifying the ΔP SR 3.7.10.4

SR 3.7.10.4 originally required a verification of the capability of a CREFS train to maintain a positive pressure of greater than or equal to 0.125 inches water gauge relative to the outside atmosphere. SNC proposed modifying this SR into a verification that the CRE ΔP is within the limits in the CRIP. With this change the requirement becomes the verification that CRE ΔP measurements are consistent with the ΔP measurements which existed at the time of the ASTM E741 test.

The performance of such measurements is a qualitative method of assessing whether the inleakage characteristics of the CRE remain the same as when the ASTM E741 test was

performed. The ΔP of the CRE should remain unchanged if: 1) the ventilation systems located within the CRE or traversing the CRE perform in a manner consistent with their performance during the ASTM E741 testing; 2) ventilation systems located in areas adjacent to the CRE or traversing areas adjacent to the CRE perform in a manner consistent with their performance during the ASTM E741 testing; and 3) there is no degradation in the CRE.

The licensee proposed to perform this surveillance on a 24-month basis. It necessitates an assessment of the control room ventilation systems performance and may necessitate an assessment of those ventilation systems affecting the CRE or those areas adjacent to the CRE. If these ventilation systems perform in a manner different than they performed during the ASTM E741 test, it is likely that different ΔP values will result. In that case, an assessment will need to be made whether the change in ΔP represents a degradation in integrity. If there is a degradation, the assessment needs to be made whether the licensing basis is still met even though there has been a degradation.

The proposed change addresses some of the inherent weaknesses associated with the existing ΔP surveillance. It is not based upon one measurement. The measurement is made with respect to adjacent areas. It incorporates the flow rates of the control room ventilation systems so that flow balances are performed which identify possible inleakage sources. The measurements also recognize the dynamic nature of ventilation systems and the impact of adjacent areas on the CRE. The NRC staff finds this proposed change acceptable.

3.2.12 Adding SR 3.7.10.5 to verify CRE integrity

SNC proposed adding a new SR which verifies CRE integrity in accordance with the CRIP. This SR adds the integrity aspect of the CRE to the TS.

The CRIP is a more forthright manner of determining CRE integrity. The previous method, the ΔP test, did not provide an accurate determination and was too limited in its application. Therefore, the proposed change is acceptable.

3.2.13 Adding the CRIP to the Programs and Manuals portion of the TSs

The licensee proposed that a new section, 5.5.18, “Control Room Integrity Program (CRIP),” be added to the Programs and Manuals Section of the TSs. The program is being added in conjunction with the licensee’s utilization of various aspects of RG 1.196.

The CRIP is established and implemented to ensure the integrity of the CRE. It is intended to demonstrate that the licensing basis remains valid given a radiological, hazardous chemical or fire event. The CRIP is intended to provide the facility the ability to conclude that the radiological doses of GDC 19 would still be met in the event of a radiological accident and that control room operators will be able to maintain reactor control from either the main control room or from the alternate shutdown panels in the event of a hazardous chemical or fire challenge. In order to provide such a demonstration, the CRE must be tested.

The frequency at which the CRE will be tested is specified in TS 5.5.18 and the CRIP. TS 5.5.18 refers to RG 1.197 for the frequency. RG 1.197 also details the test methodology. Furthermore, both RG 1.197 and TS 5.5.18 indicate that testing should also be performed when

changes are made to structures, systems, and components (SSCs) that could impact CRE integrity. TS 5.5.18 accounts for the fact that the SSCs that may affect CRE integrity may be either internal or external to the CRE. Additionally, modifications and repairs to the CRE will be ongoing. Associated with such modifications and repairs will be the necessity to conduct CRE inleakage testing following such actions. TS 5.5.18 addresses this necessity, and it also indicates that testing should be commensurate with the type and degree of modification or repair. TS 5.5.18 has incorporated the requirement to test the CRE if the conditions associated with a particular challenge result in a change in operating mode, system alignment, or system response that could result in a new limiting condition for inleakage. Testing is to be conducted in the alignment that results in the greatest consequence to the operators.

The NRC staff recognizes that the application of any test standard such as ASTM E741 to any application may necessitate the taking of certain exceptions to the test protocol. The exceptions may be taken for a variety of reasons. These exceptions are usually necessitated because of the inability to perform certain aspects of the test or to perform the test in a certain manner. Other examples include that the particular aspect of the protocol does not apply or the aspect is covered by another part of the protocol. However, in no case should the intent of the exception be to limit the ability to obtain legitimate results.

It is intended that any testing of the CRE represent the CRE's inleakage characteristics with the control room and other ventilation systems located within or serving the CRE or adjacent areas functioning in a manner consistent with the facility's licensing basis for each challenge. It must be recognized that there may be more than one mode of operation for similar types of challenges. For example, the control room ventilation systems may be aligned for an engineered safety features system to operate in one manner for a radiological challenge which results in a safety injection (SI) whereas they would operate in another manner if an SI signal is not generated. In a similar fashion, the manner in which a facility's ventilation systems respond to a hazardous chemical challenge involving chlorine may be different than the response to a sulfuric acid challenge. The manner in which the control room ventilation systems operate may be different depending upon the location of the fire (internal to the CRE versus external to the CRE).

The licensee has included, as part of it's CRIP, demonstrations that CRE inleakage is less than 43 cfm when the control room ventilation systems are aligned in their pressurization mode of operation; less than 600 cfm when the control room ventilation systems are aligned in the isolation mode of operation; and less than 2340 cfm when the control room ventilation systems aligned in their normal mode of operation. These demonstrations represent the inleakage characteristics of the FNP control room ventilation systems operating in the manner that would be expected in the event of a radiological, hazardous chemical or fire challenge. The licensee proposed that these demonstrations be performed using RG 1.197 and ASTM E741.

SNC also proposed that a demonstration be performed which shows that the leakage characteristics of the CRE will not result in the simultaneous loss of reactor control capability from the control room and the hot shutdown panels. This demonstration is intended to ensure that a fire located internal to or external to the CRE will not propagate such that either it or it's associated byproducts will render both the control room and the hot shutdown panels incapable of maintaining reactor control. Such a demonstration may be qualitative in nature but may incorporate the leakage characteristics of the CRE including those ventilation systems which may be functioning to ameliorate such conditions.

The licensee proposed that the CRIP include a CRE configuration control and design and licensing bases control program and a preventative maintenance program. These programs would contain, at a minimum, a determination of whether the CRE differential pressure, relative to adjacent areas, and the control room ventilation system flow rate values are consistent with their values at the time the ASTM E741 test was performed. If these values have changed, the licensee must determine how this change has affected the inleakage characteristics of the CRE. If there has been a degradation in the inleakage characteristics of the CRE since the ASTM E741 test, then a determination will be made whether the licensing-basis analyses remain valid. If the licensing-basis analysis remains valid, then the CRE is OPERABLE provided Items a, b and d of TS 5.5.18 are also met.

Finally, the CRIP contains the manner in which the CRE will be tested and the associated test frequencies. Reference is made to RG 1.197.

The CRIP represents the manner in which the licensee will demonstrate CRE integrity. SNC has proposed a CRIP which incorporates the FNP design and licensing-basis details. It has been formulated following numerous discussions with the NRC staff. The proposal reflects the evolution of the NRC staff's guidance on the CRIP from that which was presented in RG 1.196. This guidance is current as of the date this SE was issued. The NRC staff finds the proposed TS 5.5.18 to be acceptable.

3.2.14 Modifying the testing frequency of the VFTP

The licensee proposed that the frequency of testing associated with the VFTP be modified to the frequency of Revision 3 to RG 1.52. Presently, the testing frequency of the VFTP is in accordance with Revision 2 to RG 1.52. The NRC staff finds the proposed change acceptable.

3.3 Technical Evaluation - Summary and Conclusions

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impacts of the proposed change to the FNP TSs. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0, above. The NRC staff compared the doses estimated by SNC to the applicable criteria identified in Section 2.0 and finds, with reasonable assurance, that the licensee's estimates of the control room doses will continue to comply with these criteria (5 rem whole body or 30 rem thyroid). The NRC staff also finds, with reasonable assurance, that the licensee's estimates of the EAB and LPZ will continue to comply with these criteria (25 rem whole body and 300 rem thyroid). Therefore, the proposed license amendment is acceptable with regard to the radiological consequences of postulated DBAs. Furthermore, the licensee appropriately incorporated the guidance of RG 1.196 into the TSs. Therefore, the NRC staff finds the proposed TS changes to be acceptable.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations, if operation of the facility, in accordance with the amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the

possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The amendment has been evaluated against the three standards in 10 CFR 50.92(c). In its analysis of the issue of no significant hazards consideration, as required by 10 CFR 50.91(a), the licensee has provided the following:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. This is a revision to the TS for the control room ventilation system which is a mitigation system designed to minimize inleakage and to filter the control room atmosphere to protect the operator following accidents previously analyzed. An important part of the system is the control room envelope (CRE). The CRE integrity is not an initiator or precursor to any accident previously evaluated.

   Editorial changes and implementation of the guidance in Regulatory Guide 1.52, Revision 3 for testing cannot be initiators of any accident. Therefore, the probability of any accident previously evaluated is not increased. Performing tests and implementing programs that verify the integrity of the CRE and control room habitability ensure mitigation features are capable of performing the assumed function. Therefore, the consequences of any accident previously evaluated are not increased.

   Therefore, it is concluded that this change does not significantly increase the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The changes will not alter the requirements of the control room ventilation system or its function during accident conditions. No new or different accidents result from performing the new or revised actions and surveillances or programs required. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by these changes. The proposed changes will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without mitigating actions. The proposed changes do not affect systems that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition.

Therefore, it is concluded that this change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis, and based on this review, has determined that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff finds that the amendment request involves no significant hazards consideration.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Alabama official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change the surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (69 FR 53095). Additionally, the Commission has made a final no significant hazards consideration with respect to this amendment. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

8.0 REFERENCES

1. Letter from L. M. Stinson, SNC to NRC, “Joseph M. Farley Nuclear Plant Units 1 and 2, Request to Revise Technical Specifications, Control Room Habitability,“ August 25, 2004.

2. Letter from L. M. Stinson, SNC to NRC, “Joseph M. Farley Nuclear Plant Units 1 and 2, Request to Revise Technical Specifications, Control Room Habitability,“ September 27, 2004.

3. Letter from L. M. Stinson, SNC to NRC, “Joseph M. Farley Nuclear Plant Units 1 and 2, Request to Revise Technical Specifications, Containment Equipment Hatch,“ August 29, 2003.

4. Letter from L. M. Stinson, SNC to NRC, “Joseph M. Farley Nuclear Plant, Response to Generic Letter 2003-01 - Control Room Habitability,“ August 25, 2004.

5. Letter from S. Peters, NRC to L. M. Stinson, SNC, “Joseph M. Farley Nuclear Plant, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MC0625 AND MC0626),” September 30, 2004.

6. Letter from J. Zimmerman, NRC to D. N. Morey, SNC, “Issuance of Amendments, Joseph M. Farley Nuclear Plant, Units 1 and 2 (TAC Nos. M98120 and M98121),” April 29, 1998.

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Table 1 (page 1 of 3)
Parameters and Assumptions Used in Analysis of Post-LOCA Doses

Core thermal power	2,831 MWt (2,775 x 1.02)
Containment (CTMT) free volume	2.03×10^6 ft^3
Volume fractions	
Sprayed (SPR)	0.822
Unsprayed (UNSPR)	0.178
Mixing rate between SPR and UNSPR CTMT volumes	12,000 ft^3/min
Core fission product inventories	See FSAR Table 15.1-4
Activity released to CTMT	
Noble gases	100 % of core inventory
Iodines	50 % of core inventory
Plateout of elemental iodine activity released to CTMT	2.7 h^{-1}
Form of iodine activity in CTMT available for release	
Elemental	95.5 %
Organic	2.0 %
Particulate	2.5 %
Spray removal constants	
Elemental	10 h^{-1} (DF < 21)
	0.0 h^{-1} (DF $\geq$ 21)
Methyl	0.0
Particulate	5.4 h^{-1} (Injection)
	5.0 h^{-1} (Recirc, DF < 50)
	0.5 h^{-1} (DF > 50, t < 8 hr)
	0.0 h^{-1} (t > 8 hr)

Table 1 (page 2 of 3)
Parameters and Assumptions Used in Analysis of Post-LOCA Doses

Time to reach decontamination factor	
Elemental	24 min
Methyl	N/A
Particulate	8 h
Containment leak rate	
0-24 h	0.15 %/day
1-30 days	0.075 %/day
Emergency core cooling system leakage	
50% of the core is released to the sump	
Leakage starts at 20 minutes after the start of the accident	
Penetration Room Filtration is manually started 30 minutes after the start of the accident	
Flashing Fraction (0 to 1.3 hours)	15%
Flashing Fraction (1.3 to 720 hours)	10%
Penetration Room Filtration (elemental & organic)	89.5%
Leakage	4000 cc/hr
Calculation assumes leakage is doubled	
Filtered pressurization rate	375 ft^3/min
Filtered recirculation rate	2,700 ft^3/min
Unfiltered inleakage rate	53 ft^3/min
Filter efficiencies (all forms of iodine)	
(Note: Filter efficiencies have been reduced by 0.5% for all forms of iodine to account for bypass leakage)	
Pressurization air	98.5 %
Recirculation air	94.5 %
Volume of Control Room	114,000 ft^3

Table 1 (page 3 of 3)
Parameters and Assumptions Used in Analysis of Post-LOCA Doses

Operator breathing rate	3.47×10^{-4} m^3/s
Percent of time operator is in control room following LOCA	
0-1 day	100 %
1-4 days	60 %
4-30 days	40 %

Table 2
CR Atmospheric Dispersion Factors

Time Interval	χ/Q Value (sec/m^3)
0-2 hours	1.66×10^{-3}
2-8 hours	1.38×10^{-3}
8-24 hours	7.20×10^{-4}
1-4 days	5.60×10^{-4}
4-30 days	4.21×10^{-4}

TABLE 3
EAB Atmospheric Dispersion Factors

Time Interval	χ/Q Value (sec/m^3)
0-2 hrs	7.6×10^{-4}

TABLE 4
LPZ Atmospheric Dispersion Factors

Time Interval	χ/Q Value (sec/m^3)
0-2 hours	2.8×10^{-4}
2-8 hours	1.1×10^{-4}
8-24 hours	1.0×10^{-5}
1-4 days	5.4×10^{-6}
4-30 days	2.9×10^{-6}