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. MC0625 AND MC0626)

Dear Mr. Stinson:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 165 to Facility Operating License No. NPF-2 and Amendment No. 157 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 in response to your application dated August 29, 2003, as supplemented by letters dated November 11, 2003, and May 5, June 10, August 5, August 25, and September 27, 2004.

The amendments revise Technical Specifications Limiting Condition of Operation 3.9.3, "Containment Penetrations." The proposed changes would allow the equipment hatch to be open during core alterations and/or during movement of irradiated fuel assemblies within containment.

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. 165 to NPF-2
2. Amendment No. 157 to NPF-8
3. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page
<u>DISTRIBUTION:</u> See next page
DOCUMENT NAME: G:\DLPM\PDII-1\Farley\
Package Number: ML
Amendment Number: ML042820368
Tech Spec Number: ML *No Significant Changes to SE

OFFICE	PDII-1/PM	PDII-1/LA (A)	DSSA/SC	IPSB/BC*	IROB/SC	OGC	PDII-1/SC (A)
NAME	SPeters	DClarke	RDennig	DThatcher for TQuay	TBoyce	DFruchter	MRoss-Lee
DATE	09/30/04	9/30/04	09/29/04	09/29/04	9/30/04	09/29/04	9/30/04

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SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MC0625 AND MC0626)

Date: September 30, 2004

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EHackett
MRoss-Lee
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JHayes
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RHarvey
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RSchmitt
PHearn
CHarbuck
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GHill (4)
TBoyce
WBeckner
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DLPMDPR

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. 165
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), and May 5, June 10, August 5, August 25, and 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. 165, 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. 157
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 29, 2003, as supplemented by letters dated November 11, 2003, and May 5, June 10, August 5, August 25, and 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. 157, 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. 165

TO FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

ATTACHMENT TO LICENSE AMENDMENT NO. 157

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
3.9.3-1	3.9.3-1
3.9.3-2	3.9.3-2
B 3.9.3-1	B 3.9.3-1
B 3.9.3-3	B 3.9.3-3
B 3.9.3-4	B 3.9.3-4
B 3.9.3-5	B 3.9.3-5
----	B 3.9.3-6

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 165 TO FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 157 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 29, 2003 (Ref. 1), as supplemented by letters dated November 11, 2003, and May 5, June 10, August 5, August 25, and September 27, 2004 (Refs. 2 through 7), the Southern Nuclear Operating Company, Inc. (SNC) et al., submitted a request for changes to the Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2, Technical Specifications (TSs). The requested changes would revise TS Limiting Condition of Operation (LCO) 3.9.3, "Containment Penetrations." The proposed changes would incorporate the radiological assumptions in Regulatory Guide (RG) 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," for the fuel-handling accident (FHA) to allow for the containment equipment hatch to be open during core alterations and/or during movement of irradiated fuel assemblies. The November 11, 2003 and May 5, June 10, August 5, August 25, and September 27, 2004, letters provided clarifying information that did not change the scope of the license amendment request (LAR) as originally noticed, and did not change the NRC staff's initial proposed no significant hazards consideration as published in the *Federal Register*.

This LAR is related to a LAR dated August 25, 2004 (Ref. 8), in which SNC proposed to incorporate the control room habitability guidance of RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors." 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

2.1 Offsite and Control Room Dose

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. 10 CFR 100.11, “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. The regulatory requirements from which the NRC staff based its review on are contained in in 10 CFR Part 50, Appendix A, GDC 19 and 10 CFR 100.11, as supplemented by Regulatory Position 4.4 and 4.5 of RG 1.195[1]. Except where the licensee proposed a suitable alternative, the NRC staff used the regulatory guidance provided in the following documents in performing this review.

- RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants.”
- RG 1.195 (For the FHA only)
- RG 1.196

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 25, 2004, LAR (Ref. 8), as supplemented.

2.2 Emergency Worker Dose

In the event of an FHA resulting in damage to irradiated fuel, FNP has proposed to have a designated, trained crew of workers available to shut the containment equipment hatch within 60 minutes after notification and direction from the plant control room. To ensure that the licensee has a means for controlling radiological exposure to the emergency workers, during an emergency, the NRC staff evaluated the proposal per the requirements in 10 CFR 50.47(b)(11), “Emergency Plans.”

The regulatory guidance from which the NRC staff based its acceptance of emergency worker exposure are as follows:

- EPA 400-R-92-001, May, 1992, “Manual of Protective Action Guides and Protective Actions for Nuclear Incidents”
- RG 1.195

3.0 TECHNICAL EVALUATION

SNC performed reanalyses of the FHA described in the FNP UFSAR Chapter 15 accidents. Such reanalyses were required because of the licensee’s proposal to: 1) increase the amount

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[1]These acceptance criteria are applied to only the FHA.

of unfiltered inleakage into the control room; 2) change the release characteristics from the containment; and 3) change the atmospheric dispersion factors used. These changes would alter the releases from the FHA, and the offsite and control room doses.

3.1 FHA Radiological Consequence Analysis

This accident analysis postulates that the spent fuel assembly with the highest inventory of fission products of the 157 assemblies in the core is dropped during refueling. All of the fuel rods in the assembly are conservatively assumed to rupture, releasing the radionuclides within the fuel rod to the reactor cavity water. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod clad. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released because of the accident. Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity depending on their physical and chemical form. The licensee assumed no decontamination for noble gases, an overall effective decontamination factor of 200 for radioiodines, and retention of all particulate fission products. SNC also assumed that essentially 100 percent of the fission products released from the reactor cavity are released to the environment in 2 hours without any credit for filtration.

The assumptions that were found acceptable to the NRC staff are presented in Table 1, and the exclusion area boundary (EAB), LPZ, and control room doses estimated by the licensee for the FHA were found to be acceptable. The NRC staff performed independent calculations and confirmed the licensee's conclusions.

3.1.1 Atmospheric Dispersion Estimates

3.1.1.1 Meteorological Data

The licensee 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) for analyzing the in-containment FHA for this LAR. The resulting control room χ/Q values represent a change from those χ/Q values used in the current UFSAR analyses. The only new χ/Q values generated for this LAR 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 for NRC staff review 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 of the 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. The 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 performed further review using computer spreadsheets. 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. The 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 FNP UFSAR Table 2.3-8B shows an apparent lack of southerly winds in the 2000 through 2003 data set. The licensee 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 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 design-basis accident assessments for the purposes of this LAR.

3.1.1.2 Control Room Atmospheric Dispersion Factors

The licensee calculated the 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 each of the two (Units 1 and 2) containment hatch door releases to each of the two (Units 1 and 2) control room emergency air intakes, and SNC used the resulting bounding (highest) χ/Q values 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:

- SNC modeled the containment hatch door releases as a ground-level point source. The licensee took into consideration the difference in elevation between the release height and the intake height.

- The containment hatch doors are generally located on the opposite side of the reactor buildings with respect to the control room emergency air intakes. In order for the hatch door releases to reach these intakes, the effluents would need to travel around the reactor building. Nonetheless, the licensee used straight-line horizontal distances between the hatch doors and the air intakes in the dispersion modeling. This assumption is conservative, which yields shorter distances between the release points and the control room emergency air intakes as compared to traveling around the buildings.

- The licensee only considered the portion of the reactor building that is higher than the containment hatch doors in the building wake analysis. This approach is conservative because smaller building cross-sectional areas produce higher χ/Q values.

The resulting control room emergency air intake χ/Q values were used to analyze: (1) routine outside air makeup through the control room normal air intake prior to control room isolation; (2) unfiltered inleakage when the control room is isolated but not pressurized; and (3) emergency filtered pressurization air makeup through the control room emergency air intake when the control room is isolated and pressurized. Although the normal air intake is closer to the containment hatch door release pathways as compared to the emergency air intakes, the routine outside air makeup through the normal air intake can still be represented by control room emergency air intake χ/Q values for the FHA. This can be done because the licensee derived the control room emergency air intake χ/Q values using the straight-line distances from the equipment hatch doors through the containment structure rather than the actual distances on the travel paths around the containment to the emergency air intakes. This assumption results in conservatively short distances that bound the actual plume travel distances between the equipment hatch doors and the control room normal air intake.

To address the 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 one-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 FNP, Unit 1 containment hatch door releases to the FNP, Unit 1 control room intake. The resulting 0–2 hour control room χ/Q value increased 4.8 percent during the critical period of the FHA (0–2 hours). The resulting control room χ/Q value is presented in Table 2.

Note that the licensee's original submittal indicated that both 0–2 hour and 2–8 hour χ/Q values were used in evaluating control room doses for the FHA analysis. However, in the answer to NRC Question 2 contained in the response to the request for additional information (RAI) letter dated August 5, 2004, SNC stated that χ/Q values beyond two hours have no impact on the dose consequences since the release of activity from the containment is essentially complete after two hours. The NRC staff concurs with this assessment and, as such, the 2–8 hour control room χ/Q value was not reviewed by the NRC staff as part of this LAR.

By letter dated November 11, 2003, the licensee also presented information concerning control room and technical support center (TSC) χ/Q values associated with releases from the reactor buildings and vent stacks as well as TSC χ/Q values associated with releases from the containment hatch doors in response to an NRC staff RAI. These χ/Q values were not used to support this licensing action; instead, they are intended for use by the licensee in future LARs. Consequently, the NRC staff did not review these χ/Q values as part of this LAR.

In summary, the NRC staff has 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 0–2 hour CR χ/Q value is presented in Table 2. On the basis of this review, the NRC staff concludes that this χ/Q value is acceptable for use in performing control room dose assessments for an in-containment FHA.

3.1.1.3 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. 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 this LAR for EAB and LPZ dose assessments.

3.1.2 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. 9). 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, 33 cfm during the isolation mode, and 87 cfm during the normal alignment. All of these modes are used for the FHA analysis. The proposed values assumed for the FHA are provided in Table 1. These values plus 10 cfm for ingress and egress are conservative compared to the measured values.

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 would be habitable during the FHA and that this amendment may be approved before the final resolution of the generic issue. The NRC staff bases 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 the licensee's unfiltered inleakage assumption for the purpose of this LAR 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.

3.1.3 Offsite Doses

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

3.2 <u>Emergency Worker Exposure</u>

The proposed revision to TS 3.9.3 would allow for the containment equipment hatch to be open during core alterations and/or during movement of irradiated fuel assemblies within containment. The licensee evaluated the overall impact that the proposed modification would have on its ability to control radiation exposure to emergency workers, in the event of an FHA inside containment. Additionally, procedures are in place to ensure that exposures are controlled consistent with Environmental Protection Agency Emergency Worker and Lifesaving

Activity Protective Action Guides as referenced in 10 CFR 50.47(b)(11). Maintenance personnel have procedural guidance for normal closure of the containment equipment hatch and perform the activity routinely during refueling outages. Also, training and actual performance or simulation of the hatch closure is required of Mechanical Maintenance Journeymen.

The licensee indicated that a pre-job brief would be performed with the designated trained hatch closure crew prior to starting core altercations. The pre-job brief would discuss: the requirements of the Radiation Worker Permit or Health Physics Plan that would be used for the closure of the containment equipment hatch, expected radiological conditions, protective clothing requirements and actions to take if the plant emergency alarm is used.

The plant procedures require personnel to wear dosimetry when entering a Radiologically Controlled Area in support of the containment equipment hatch closure. Protective clothing would be required to be worn in contaminated areas and authorization would be given to allow personnel to wear protective clothing over their personal clothing. Eye protection would also be required.

Through General Employee Training, maintenance personnel train in the use of respiratory devices or other methods to limit the intake of radioactive material. While respirators are available, they would not be required for entries to provide emergency closure of the hatch. Although not anticipated, if a person were to be potentially exposed to airborne radioactive iodine, the issuance of potassium iodide (KI) as a thyroid blocking agent would be considered. Existing plant procedures are in place to provide guidance for use of respiratory equipment or utilization of KI should the need arise. Because of the existing training, no additional training would be required to support closure activities.

If any monitor alarms as a result of an FHA, Health Physics (HP) would perform the following:

(1) safely evacuate personnel;

(2) contact the control room and HP Supervisor for additional actions;

(3) secure access to the area by non-essential personnel;

(4) conduct additional sampling as directed; and

(5) provide dedicated HP support to the designated trained hatch closure crew to include escort to the work area, setting dose rates for the workers and escort out of containment.

The radiological conditions at the containment equipment hatch would be assessed directly by HP personnel providing support to the designated trained hatch closure crew using an instrument that can detect high levels of Beta radiation and through the following air sampling activities:

(1) A Continuous Air Monitor would be in service inside containment and the area of the containment equipment hatch outside of containment any time the hatch is open and fuel movement is in progress. Short interruption of monitoring to support response check of the instrument, filter change out or replacement of a malfunctioning instrument with one on standby, is expected.

(2) A Low Volume Air sampler would be running continuously inside containment and the area of the equipment hatch outside of containment any time the equipment hatch is open and fuel movement is in progress. Air sampling would be for particulate and iodine activity.

(3) If the situation warrants, noble gas samples would be taken.

In the event that contamination of personnel occurs, whole body counts would be utilized to assess the radiological exposure. If additional assessments are required due to known or suspected intakes of radioactive material, follow-up bioassay sampling and analysis may be conducted.

The licensee performed an evaluation of potential doses to the worker's thyroid, whole body, and skin. The dose to a crew member inside the containment for 1 hour would be approximately 46.1 rem thyroid, 1.1 rem skin, and <0.1 rem whole body. Crew members transiting to the containment equipment hatch from inside the containment would include their transit time in the 1 hour total stay time. Crew members transiting to the hatch from outside the containment would be exposed to activity exhausted from the containment (which would reduce the exposure inside containment). Assuming complete exhaust during transit, the resultant doses would be about 1.3 rem thyroid, and <0.1 rem whole body and skin. Radiation doses of these magnitudes are well within regulatory exposure limits and do not represent an impact to worker health.

No other significant contributions to worker dose are expected since the containment purge filter is outside the containment and the pre-access filter, typically not used during refueling, is on the opposite side of the containment and is partially shielded by the steam generator and compartment walls.

Based on the above, the NRC staff finds that there is reasonable assurance that the licensee would be able to control radiation doses to its emergency workers, within the regulatory limits.

3.3 Technical Specification Changes

The licensee's current LCO 3.9.3.a required the equipment hatch to remain closed during core alterations and movement of irradiated fuel assemblies within containment by stating the equipment hatch shall be "...closed and held in place by four bolts;". To allow for the hatch to remain open during these conditions, the licensee proposed modifying the LCO to state that the equipment hatch, "... is capable of being closed and held in place by four bolts;". This change is supported by the licensee's analysis and by the NRC staff's evaluation. Therefore, the NRC staff finds the proposed change acceptable.

In addition, the licensee proposed adding SR 3.9.3.3. This SR would require that the licensee verify the capability to install the open equipment hatch every 7 days. The NRC staff finds that this surveillance and its associated period would verify that the assumptions for a 1 hour containment equipment hatch closure remain valid. Therefore, the NRC staff finds the proposed change acceptable.

3.4 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. Based on its review, 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 finds, with reasonable assurance, that the licensee's estimates of the control room doses would continue to comply with these criteria (5 rem whole body or equivalent). The NRC staff also finds, with reasonable assurance, that the licensee's estimates of the EAB and LPZ would continue to comply with these criteria (6.3 rem whole body and 75 rem thyroid). Therefore, the proposed license amendment is acceptable with regard to the radiological consequences of the postulated FHA. Furthermore, NRC staff also finds that there is reasonable assurance that the licensee will be able to control radiation doses to its emergency workers, within the regulatory limits.

4.0 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.

5.0 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 (68 FR 64137). 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.

6.0 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.

7.0 REFERENCES

1. Letter from J. B. Beasley, Jr., SNC to U.S. NRC, "Joseph M. Farley Nuclear Plant Units 1 and 2, Request to Revise Technical Specifications, Containment Equipment Hatch," August 29, 2003.

2. Letter from J. B. Beasley, Jr., SNC to U.S. NRC, “Joseph M. Farley Nuclear Plant Units 1 and 2, Response to Request for Additional Information Related to Request to Revise Technical Specifications - Containment Equipment Hatch,“ November 11, 2003.

3. Letter from L. M. Stinson, SNC to U.S. NRC, “Joseph M. Farley Nuclear Plant Units 1 and 2, Response to Second Request for Additional Information Related to Reques to Revise Technical Specifications - Containment Equipment Hatch,“ May 5, 2004.

4. Letter from L. M. Stinson, SNC to U.S. NRC, “Joseph M. Farley Nuclear Plant Units 1 and 2, Revised Response to Selected Questions from Second NRC Request for Additional Information Related to Request to Revise Technical Specifications - Containment Equipment Hatch,“ June 10, 2004.

5. Letter from L. M. Stinson, SNC to U.S. NRC, “Joseph M. Farley Nuclear Plant Units 1 and 2, Response to Third Request for Additional Information Related to Request to Revise Technical Specifications - Containment Equipment Hatch,“ August 5, 2004.

6. Letter from L. M. Stinson, SNC to U.S. NRC, “Joseph M. Farley Nuclear Plant Units 1 and 2, Response to Fourth Request for Additional Information Related to Request to Revise Technical Specifications - Containment Equipment Hatch,“ August 25, 2004.

7. Letter from L. M. Stinson, SNC to U.S. NRC, “Joseph M. Farley Nuclear Plant Units 1 and 2, Response to Request for Additional Information Related to Request to Revise Technical Specifications - Containment Equipment Hatch,“ September 27, 2004.

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

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

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Date: September 30, 2004

Table 1 (sheet 1 of 2)
Parameters and Assumptions Used in Analysis of FHA
(Accident in Containment with Equipment Hatch Open)

Core thermal power	2831 Mwt
Time between plant shutdown and accident	100 h
Minimum water depth between tops of Damaged fuel rods and water surface	23 ft
Damage to fuel assembly	All rods ruptured
Fuel assembly activity	Highest powered fuel assembly in core region discharged
Activity release from assembly	Gap activity in ruptured rods per RG 1.195, Table 2
Radial peaking factor	1.7
Decontamination factor in water	
Elemental iodine (99.75%)	400
Organic iodine (0.25%)	1
Noble gases	1
Exhaust flow rate	53,500 cfm
Exhaust isolation time	N/A
Iodine filtration system	Containment purge system (not credited)
Filter efficiency (all species)	N/A
Dose Conversion Factors	ICRP 30

Table 1 (sheet 2 of 2)
Control Room Parameters Used in Analysis of FHA
(Accident in Containment with Equipment Hatch Open)

Normal HVAC unfiltered intake (ft^3/min)	3000
Unpressurized unfiltered infiltration (ft^3/min)	600
Filtered pressurization rate (ft^3/min)	450
Pressurized unfiltered infiltration (ft^3/min)	450
Filtered recirculation rate (ft^3/min)	2700
Unfiltered ingress/egress rate (ft^3/min)	10
Filter efficiencies (all forms of iodine) (%)	
Pressurization air	98.5[(1)]
Recirculation air	94.5[(1)]
Volume (ft^3)	114,000
Operator breathing rate (m^3/s)	3.47×10^{-4}
Percent of time operator is in control room following loss-of-coolant accident	
0-1 day	100
1-4 days	60
4-30 days	40

Notes

(1) Filter efficiencies have been reduced by 0.5 percent for all forms of iodine to account for bypass leakage.

TABLE 2
CR Atmospheric Dispersion Factor

Time Interval	χ/Q Value (sec/m^3)
0–2 hrs	8.79×10^{-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 hrs	2.8×10^{-4}

Joseph M. Farley Nuclear Plant, Units 1 & 2

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