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SOUTHERN COMPANY
Energy to Serve Your World℠

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Joseph M. Farley Nuclear Plant Units 1 and 2
Response to Second Request for Additional Information Related to
Request to Revise Technical Specifications - Containment Equipment Hatch

Ladies and Gentlemen:

By letter dated August 29, 2003, Southern Nuclear Operating Company (SNC) submitted a request to amend the Farley Nuclear Plant (FNP) Unit 1 and Unit 2 Technical Specifications (TS), to allow the equipment hatch to be open during core alterations and/or during movement of irradiated fuel assemblies within containment. By letter dated November 11, 2003, SNC submitted a response to a Request for Additional Information related to that submittal. NRC letter dated May 3, 2004 requested SNC to provide additional information related to the request. Enclosure 1 provides the NRC questions and the SNC responses. Enclosure 2 provides portions of the current fuel handling calculation related to the evaluation of the open equipment hatch. Enclosure 3 is a Compact Disc (CD) containing the meteorological data files used as direct input to the ARCON96 computer code, the output of the ARCON96 computer code, and the calculation of χ/Q.

(Affirmation and signature are on the following page).

A001

Mr. L. M. Stinson states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

This letter contains no NRC commitments. If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

L. M. Stinson

Sworn to and subscribed before me this 5th *day of* May*, 2004.*

Notary Public

My commission expires: 6-7-05

LMS/WAS/sdl

Enclosure 1 – Response to NRC Request for Additional Information
Enclosure 2 – Excerpt from Design Bases Fuel Handling Calculation
Enclosure 3 – ARCON96 Input Data Files, ARCON96 Output Data Files, χ/Q Calculation

cc: Southern Nuclear Operating Company
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U. S. Nuclear Regulatory Commission
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Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer

ENCLOSURE 1

Joseph M. Farley Nuclear Plant Units 1 and 2
Response to Second Request for Additional Information Related to
Request to Revise Technical Specifications - Containment Equipment Hatch

<u>SNC Response to NRC Request for Additional Information</u>

NRC Question 1

Please provide the design-bases parameters, assumptions, and methodologies (other than those provided in the August 29, 2003, submittal) that were changed in the radiological design-basis accident analyses as a result of the proposed TS change. Also, please provide justification for the changes. If there are many changes it would be helpful to compare and contrast them in a table.

SNC Response:

No changes beyond those included in the August 29, 2003 submittal were made. The following previously identified changes were made consistent with the guidance of Regulatory Guide (RG) 1.195:

a. Gap activity was changed as shown in RG 1.195, Table 2.
b. Pool decontamination was changed as shown in RG 1.195, Appendix B.
c. Credit for isolation of the containment was deleted and all activity assumed exhausted.
d. Control room χ/Qs were recalculated using ARCON96, and are discussed further below.

Although not a change to the analysis, it should be noted that the 10 cfm unfiltered inleakage flow previously provided is for ingress/egress. In addition, when the control room is isolated, but not pressurized, unfiltered inleakage is assumed to be one half the emergency pressurization flow rate, or 187.5 cfm of outside air.

NRC Question 2

Based upon a preliminary review of the fuel handling accident for the proposed TS change, the reviewer is unable to match the calculated doses. Please provide the calculation for the design bases fuel handling accident.

SNC Response:

Portions of the current fuel handling calculation related to evaluation of the open equipment hatch are provided in Enclosure 2. This calculation has not been updated with the new χ/Q values based on the new meteorological tower data set. The assumed control room inleakage value shown in the current calculation and the χ/Q input will be revised when inleakage testing is completed.

NRC Question 3

What types of hoses and cables will be allowed to pass through the open equipment hatch? What provisions will be made for the designated individual to separate these to close the air lock door while reducing hazards from these hoses and cables?

SNC Response:

Air supply hoses from outside air compressors, demineralized water supply hoses, and Containment Cooler Service Water drain hoses would be routed through the equipment hatch with isolation valves close to the hatch with quick disconnects in the hatch area. Temporary power cables would be routed from the transformer outside the Containment hatch area through the equipment hatch. The temporary power supplies would have disconnects which could be opened and would have a plug type connector in the hatch area which would allow the hatch to be closed.

Procedures are in place for refueling which require that cables and hoses be conspicuously identified and have breakable connections. The Equipment hatch and personnel air locks are inspected once every 24 hours to verify hoses and extension cords are properly identified and have breakable connections. Procedures are in place which require establishment of a Maintenance Closure Response Team (MCRT). The team must be briefed and have necessary tools and procedures staged to facilitate rapid closure.

NRC Question 4

Appendix B to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 establishes quality assurance requirements for the design, construction, and operation of those structures systems or components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Appendix B, Criterion III, "Design Control," requires that design control measures be provided for verifying or checking the adequacy of design. Appendix B, Criterion XVI, "Corrective Action," requires measures to be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, defective material and equipment, and nonconformances are promptly identified and corrected. Generic Letter (GL) 2003-01, "Control Room Habitability," addresses current issues with respect to previously assumed values of unfiltered inleakage. Generally, these issues can only be resolved by inleakage testing. In light of Appendix B requirements and GL 2003-01, provide sufficient justification to explain why a value of zero for the control room's unfiltered inleakage is appropriate for this proposed TS change request. Provide details regarding your control room, design, maintenance and assessments to justify the use of this number.

SNC Response:

FNP will perform control room envelope integrity testing using the standard test method described in American Society for Testing and Materials (ASTM) standard E741, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution." Once actual inleakage is determined, this information will be used to update the calculations for this submittal. In addition, at that time the calculation will be updated with the new χ/Q values based on the new meteorological tower data set. The control room was designed for low leakage using the technology and methods prevalent at the time of FNP licensing. The primary source of inleakage is expected to be through ductwork operated at a negative pressure (on the suction side of fans), and this ductwork is typically of welded construction, minimizing the potential for inleakage. Post-accident access to the control room would typically be through the non-rad area corridor from Unit 1 or through the Technical Support Center (TSC) from Unit 2. Inleakage from the TSC area would be filtered, and the non-rad area corridor of Unit 1 has two doors in series minimizing inflow which would be diluted by the air volume of the auxiliary building. In addition to the 10 cfm ingress/egress flow assumption; 187.5 cfm unpressurized, unfiltered inleakage is assumed during the first ten minutes until the pressurization flow is manually established. Review of the calculated releases indicates most activity is released from the open containment during this first ten minute period when the unpressurized inleakage is the dominant source. The results of the current dose calculation are approximately 11% of the 50 REM thyroid limit, indicating substantially more unfiltered inleakage can be accommodated. Based on these results, the assumed unfiltered control room inleakage for the fuel handling accident could be increased by approximately a factor of 9 and doses would remain within the acceptance criteria in Regulatory Guide 1.195. Such leakage would be well in excess of results from recent industry tracer gas tests for pressurized control rooms and the LOCA inleakage acceptance criteria to be used in the control room inleakage testing to be performed in response to Generic Letter 2003-01.

NRC Question 5

The proposed TS specifies that a "designated trained hatch closure crew" is available to close the Containment Structure Equipment Hatch Shield Doors rather than a "dedicated" crew who would have no other duties. Specify what other duties the designated crew will have and where they will be stationed relative to the air lock doors.

SNC Response:

The "designated trained hatch closure crew" could have other jobs in the main power block area (not the Service Water Intake Structure, River Water structure, or other outlying areas). The designated crew would have beepers, radios, or phones which would allow quick communications should the need for containment closure arise.

NRC Question 6

Provide a detailed account of the timing and flow rates, and filtration of the control room Heating Ventilation and Air Conditioning as it responds to the accident. Please justify the assumption that one of the two emergency control room filtration trains are operating within 10 minutes of the accident. Is this action automatic or manual?

SNC Response:

The control room is normally maintained at a slightly positive pressure by providing about 1350 cfm from the computer room Heating, Ventilation and Air Conditioning (HVAC) system, about 35% of which is outside air; however, for conservatism 100% outside air is assumed. This normal air supply is monitored for high radiation, and the setpoint is expected to be reached and generate an isolation signal within 20 seconds. For conservatism, 45 seconds isolation time is assumed, after which the control room is unpressurized with inleakage assumed to be one half the emergency pressurization rate, i.e., the unpressurized infiltration is assumed to be 187.5 cfm plus 10 cfm ingress/egress flow. This continues for ten minutes, until the operator manually initiates emergency filtered pressurization and recirculation filtration. Ten minutes is the currently assumed operator action time, and is a conservative assumption given the communication between the control room operators and the refueling floor and also the outside air intake radiation monitor alarm in the control room. Outside air flow for pressurization is 375 cfm and recirculated filtered air flow is 2700 cfm for the duration of the analysis, based on technical specification limits.

NRC Question 7

What criteria will be used to determine if closure of the containment is necessary in the event that environmental conditions could impact fuel handling? Has the impact of wind on fuel handling been evaluated (for example, reduced pool visibility due to pool surface disruption)? What steps would be taken in the event of severe weather to minimize the impact of flying debris?

SNC Response:

Procedures are in place which require that the equipment hatch be closed for two environmental conditions: 1) Tornado Warning or report of tornado, and 2) Sustained winds in excess of 74 mph predicted for the site within 24 hrs. If heavy winds are expected on plant site, procedures provide guidance for securing items which could become missiles. In addition, following closure for the extreme environmental conditions noted, an inspection for damage is to be performed and the condition of the equipment hatch evaluated prior to fuel movement.

The Senior Reactor Operator (SRO) in charge of fuel handling has ultimate responsibility for conditions related to refueling. If environmental conditions such as debris or wind were to cause a distraction, the SRO could make the decision at any time to have the equipment hatch closed. Fuel movement can only be carried out if visibility is adequate for safe transport of the fuel. A plastic curtain is hung in front of the equipment hatch during outages. This curtain provides some degree of protection from light debris and wind. In addition, a Controlled Refueling Area Boundary (CRAB) is established around the refueling cavity. This area is clearly defined by the use of a physical barrier (curtain) which will also provide some protection against flying debris and wind.

NRC Question 8

Criterion 64 of 10 CFR Part 50, Appendix A states that means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents. The proposed TS change should consider how Criterion 64 will be met in the event of a Fuel Handling Accident (FHA) with the equipment hatch open. Moreover, this information should be included as part of the Bases discussion.

Provide the bases for meeting Criterion 64 for the proposed TS change. Please confirm that your emergency planning dose assessment methodology includes the ability to assess this accident. For example, does your methodology include the capability to determine the source term, release rate out of containment, meteorology and consider feedback via field monitoring health physics survey teams? Have you evaluated the need for any special radiological monitoring or survey equipment (i.e., in-plant equipment or field team survey equipment) to evaluate the radiological conditions of this accident scenario? Will your emergency response personnel be trained to deal with this accident scenario?

SNC Response:

Farley Emergency Planning dose assessment methodology does include methods for assessing an accident release which could occur due to an open containment hatch.

Should the effluent flow direction be through the open containment hatch to the environment, field monitoring teams would be dispatched by the Emergency Plan Implementing Procedures to monitor the magnitude of the release. Current dose assessment methodology allows for the field monitoring sample results to be used to back calculate source term, projected dose, and dose rates off site. Necessary radiological survey equipment is already available and personnel are currently qualified to conduct needed monitoring activities. Should the effluent flow direction be from the environment into containment through the open hatch, permanently installed radiation monitoring systems would be utilized to calculate source term, projected dose, and dose rates off site. Necessary radiological survey equipment is already available and personnel are currently qualified to conduct needed monitoring activities.

Guidance to ensure that Criterion 64 requirements are met is included in the Emergency Plan, which is the appropriate Licensee Controlled Document for this information.

NRC Question 9

10 CFR 50.36 states that:

> A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:
>
> (B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The proposed analysis utilizes an initial condition of 100 hours of fuel decay for the FHA. The proposed TS does not provide a limiting condition of operation for this initial condition. Please justify why this decay time does not meet Criterion 2 of 10 CFR 50.36 or modify the TS to include the decay time.

SNC Response:

Prior to conversion to the Improved Technical Specifications (ITS), the Farley Technical Specifications (TS) contained an LCO related to decay time. In the conversion to the ITS, this TS was relocated to the TRM consistent with the Standard Technical Specifications (STS) for Westinghouse Plants contained in NUREG-1431, Revision 1 dated 04/07/95. This was approved by the NRC on November 30, 1999 with the issuance of the Farley ITS. The following contains the justification for that relocation which is still applicable to Farley.

The Decay Time TS requirement places a time limit on reactor subcriticality prior to the movement of irradiated fuel assemblies out of the reactor vessel. This ensures that sufficient time has elapsed for the radioactive decay of short-lived fission products and is consistent with the assumptions used in the fuel handling accident safety analysis. However, the schedule restraints associated with a normal refueling shutdown ensure the movement of irradiated fuel does not occur prior to the Decay Time TS limit of 100 hours. Refueling outage schedule restraints include RCS cooldown, depressurization, boration, removal of the reactor vessel head and upper internals, flooding the reactor cavity to the required level, as well as various required testing and maintenance activities. The activities and requirements associated with a normal refueling shutdown are inherent in the design and operating restrictions (i.e., cooldown and depressurization TS requirements, water level TS requirements, boron concentration TS requirements, and TS requirements to maintain and test equipment to ensure operability) associated with pressurized water reactors. Therefore, the requirements of this LCO are not specifically relied on to ensure the initial conditions assumed in the applicable safety analyses are met. Therefore, this specification was not included in the STS. Since the justification provided above is applicable to FNP and in order to conform as close as practicable to the STS, this specification is proposed for relocation from the FNP TS. The relocation of this specification is consistent with the content of the Westinghouse Standard Specifications contained in NUREG-1431, Rev. 1.

Decay Time is not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. Therefore, this specification does not satisfy Criterion 1.

Decay Time is an operating restriction or process variable that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. However, the practical aspects and schedule restraints of a normal refueling outage shutdown, which include physical design limitations (i.e., reactor head and upper internals removal) and TS restrictions associated with cooldown and depressurization, water level, boron concentration, as well as required equipment maintenance and testing, prevent the movement of irradiated fuel prior to the Decay Time limit of 100 hours after shutdown. Therefore, although the limit in this specification meets Criterion 2, the retention of the Decay Time specification is not necessary to ensure the initial conditions of the applicable safety analyses are met.

Decay Time is not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this specification does not satisfy Criterion 3.

Decay Time is not a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety. Therefore, this specification does not satisfy Criterion 4.

Each Technical Specification proposed for relocation has been evaluated to identify if it contains requirements that are addressed by PRA, and if addressed, to determine if the Specification is important to risk (i.e., contains constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk). Documents utilized to evaluate the risk insights relevant to the FNP Technical Specifications proposed for relocation include the generic evaluations performed by Westinghouse in WCAP-11618, "Methodically Engineered Restructured and Improved Technical Specifications, MERITS Program - Phase II Task 5, Criteria Application," November, 1987, the FNP PRA submitted in response to Generic Letter 88-20, and any other published PRA studies found to be applicable to FNP in a generic manner. The FNP Technical Specifications proposed for relocation were found not to be important to risk.

Emergency Plan Considerations

NRC Question 10

Will your Emergency Plan be updated to include an accident release through the equipment hatch? Will your Emergency Operating Procedures be updated to address the specific details needed to respond to this accident scenario?

SNC Response:

Emergency Planning dose assessment methodology already includes methods for assessing an accident release which could occur due to an open containment hatch.

- Field teams would be dispatched by procedure to determine if any unmonitored release from the containment was occurring. Results from field monitoring activities would be utilized to perform back calculations to determine offsite consequences.
- Dose assessment personnel would begin dose assessment on monitored releases from the containment utilizing automated and manual computerized dose assessment methods.

No additional procedure modifications are needed in order to ensure that radiological monitoring techniques determine the needed radiological release information necessary to ensure the health and safety of plant personnel or the public.

NRC Question 11

Will you inform the State Emergency Response personnel about this accident scenario?

SNC Response:

State Emergency Response personnel are already aware that accident scenarios exist that could result in radiological release through an open containment hatch. The State Emergency Response personnel are also aware of the methods for detection and determination of offsite consequences utilized by FNP.

Meteorological Monitoring Program:

NOTE: In the response to the first RAI contained in SNC letter dated November 11, 2003, raw data was provided to the NRC instead of actual corrected input files for ARCON96. This data had not been adjusted for use as direct input to the calculation (e.g., replacement of missing or erroneous data). Some of the NRC questions resulted due to this fact. Version 0 of the χ/Q calculation used this data adjusted for use as direct input to the calculation. SNC has reviewed the original data and agrees that inconsistencies were present. Therefore, new data was obtained for the years 2000 – 2003 and reviewed. This new data was used in version 1 of the χ/Q calculation. The resolution for the RAIs below are based on the review of the new data and version 1 of the χ/Q calculation. The ARCON96 input data based on this new raw data and the output from the ARCON96 computer code run is included as Enclosure 3 to this letter.

NRC Question 12

Confirm that the 2000 through 2002 meteorological data used in the atmospheric dispersion analyses are of high quality, representative of long term conditions, and suitable for use in the dispersion analyses. The primary intent of the following questions is to assess the overall quality of the meteorological data as collected and processed for use in the atmospheric dispersion calculations. These questions are also intended to ascertain whether there have been any changes made to the meteorological monitoring program from that currently described in the Farley Nuclear Plant, Final Safety Analysis Report Section 2.3.3.

During the period of data collection, did the measurement program meet the guidelines of Regulatory Guide (RG) 1.23, "Onsite Meteorological Programs"? Was the tower base area on natural surface (e.g., short natural vegetation) and was the tower free from obstructions (e.g., trees, structures) and micro-scale influences to ensure that the data were representative of the overall site area? In the case of possible obstructions, were trees, structures, etc. at least 10 times their height away from the meteorological tower? What types of surveillance activities were performed to ensure that data were of high quality? Were instrumentation problems and resulting questionable data identified and corrected in a timely manner? Were calibrations properly performed and systems found to be within guideline specifications? What additional data reviews were performed following data collection and prior to archival? If data abnormalities occurred, describe the abnormalities and why the data were still deemed to be adequate (a detailed response for each individual data point is not expected). Were the data compared with other site historical or regional data? If so, what were the findings?

What additional data reviews were performed prior to their input into the atmospheric dispersion calculations to ensure that conversion and reformatting to the ARCON96 format were properly performed?

SNC Response:

The meteorological measuring program at Farley Nuclear Plant (FNP) is implemented in accordance with FSAR Section 2.3.3 and Regulatory Guide 1.23. The towers (primary and back up) are free from obstructions such that the data collected is representative of the plant site. In order to ensure the accuracy of the met data, equipment calibrations are performed on a once per 6 month frequency in accordance with plant procedures. These instrument calibrations ensure met tower equipment is within the guideline specifications. Problems with met tower instrumentations are addressed via the plant work control process. Work orders indicating operability concerns with these instruments are given higher priority.

A review of the Radiation and Meteorological Data Acquisition (RMDA) met data is performed daily per plant procedures to ensure the validity of the data. The met tower equipment is physically and visually checked twice per week while the RMDA is reviewed once per week by the Chemistry group for data accuracy, consistency and equipment operability. There have been no abnormalities detected in these weekly reviews. If abnormalities were detected, then a review of backup met data would be performed to validate RMDA. The backup met tower data would serve to ensure the adequacy of the data. Typically the data are not compared with regional data or site historical since the backup met tower is available should abnormalities occur. The Joint Frequency Tables are printed and reviewed on annual bases prior to archival. More recently, the FNP met data for 2000 – 2003 were reviewed by ABS consultants with no outstanding issues.

Recent data supplied to the NRC indicated an inconsistency in some of the measured parameters. These data were taken from the plant process computer which was not in agreement with the local met tower instrumentation and/or RMDA. This plant process computer data have not previously been included as a part of met data review. A condition report has been written and addresses the problem of agreement between the met tower and RMDA to the Plant Process Computer.

For both version 0 and 1 of the χ/Q calculation, raw data were reviewed prior to their input into the dispersion modeling. The raw monthly data were in Excel spreadsheet format; therefore, they were saved as MS DOS text files and combined into annual files. Missing or bad data were re-labeled 999, 9999 or 99 for wind direction, wind speed and atmospheric stability, respectively. Finally, the ASCII annual text files were adjusted and saved to meet the ARCON96-required format.

<u>Meteorological Data</u>

NRC Question 13

Staff review indicates apparent weaknesses in the submitted 2000 through 2002 meteorological data as discussed below. Please check and amend the data files as appropriate. Provide a copy of the amended files and the basis for any remaining departures from the expected conditions and RG 1.23. Alternatively, provide a replacement set of high quality data.

a) The data files for each year contain non-consecutive (missing) date/hour sequences. In addition, May 2001 contains several records containing duplicate hourly labels. ARCON96 requires that the meteorological data files contain one record per hour. Any hour with invalid data should be represented with each data field completely filled in with 9s (for example, "999" for a wind direction field, "9999" for a wind speed field, and "99" for the stability class field).

SNC Response:

The above problems exist in the original Excel spreadsheets. Those problems were corrected in the ASCII data files that were used for ARCON96 modeling runs in Version 0 of the calculation. In Version 1 of the calculation, new meteorological data set were used, all missing data were properly labeled with 9s (e.g. 999, 9999 or 99 for wind direction, wind speed and atmospheric stability, respectively).

b) The ASIIMETDATA files contain wind direction values of 0°. If these values are intended to represent valid wind direction observations from the North, they are being misinterpreted by ARCON96 as invalid data (valid wind direction values provided as input to ARCON96 should range from 1° to 360°). In addition, there are several occurrences of negative wind direction values. All valid wind direction values should range from 1° to 360°; all invalid wind direction values should be identified by "999".

SNC Response:

Negative wind direction values were corrected and all invalid wind direction values were identified by "999" for ARCON96 modeling runs in Version 0 of the calculation. Valid 0° wind direction was not changed to 360°.

New meteorological data set were used (2000-2003) for the Version 1 calculation. Valid north wind direction was changed from 0° to 360°. For the Version 1 calculation, missing data tagged as –9999 in the raw data set were replaced with proper 9s for different meteorological parameters as required by ARCON96. A Fortran computer code FarlMet was developed to perform the final data processing in order to take care of the above issues.

c) Likewise, the ASIIMETDATA files contain wind speed values of -44700 mps. If these values are intended to represent invalid wind speed observations, they should be reset to "9999".

SNC Response:

The above issue was addressed in Version 0 of the calculation. The –44700 mps wind speed occurred only in the raw data set, not in the final data set used for the Version 0 calculation. The new meteorological data set used for the Version 1 calculation does not have this issue.

d) There are a number of hours in the RAWMETDATA files with delta-temperature values of -9999 °F that were classified as stability class A in the ASIIMETDATA files. There are also a number of hours with delta-temperature values of 100 °F that were classified as stability class G. If these values represent invalid delta-temperature observations, the associated stability class value should be set to "99".

SNC Response:

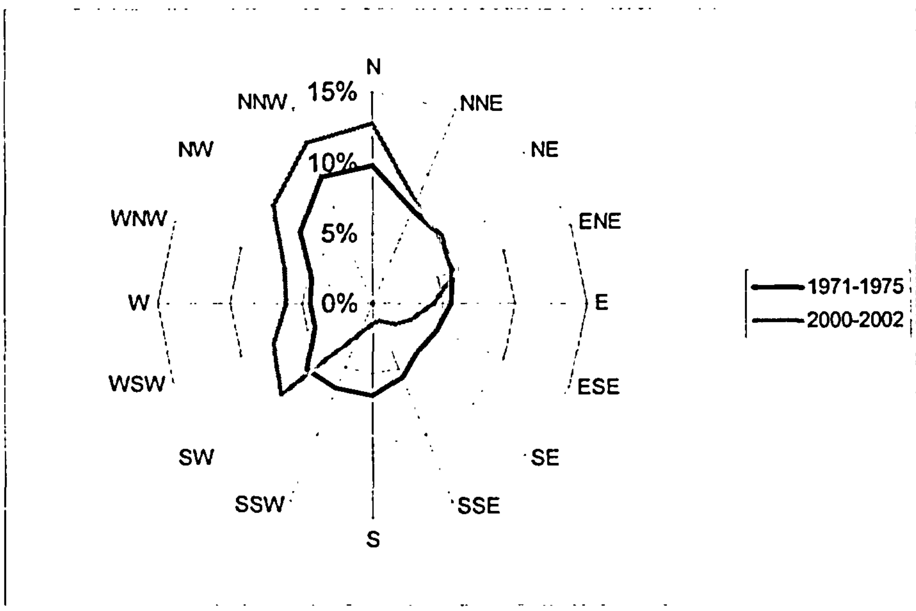
Stability classes (A – G) presented in the RAWMETDATA files were used directly for ARCON96 modeling runs in Version 0 of the calculation. Four years of new meteorological data (2000-2003) were used for Version 1 of the calculation. Questionable stability classes were labeled as "99" in the new data set.

e) A high frequency of stability class A was reported during 2002 (20.8%) as compared to an average of 2.4% for 2000 and 2001. In fact, nearly 65% of the hourly data reported from Julian day 262 (September 19, 2002) through the end of 2002 were classified as stability class A. One continuous period of stability class A data which began on hour 22 on Julian day 295 (October 22, 2002) lasted for 188 hours. Please explain what might have caused this high occurrence of stability class A conditions during the last four months of 2002.

SNC Response:

Stability classes (A-G) presented in the RAMETDATA files were used directly for ARCON96 modeling runs in Version 0 of the calculation. The new meteorological data set (2000-2003) used for Version 1 of the calculation does not have the above issue.

f) A wind rose comparing wind direction frequency distributions between the Jan 2000 - Dec 2002 onsite meteorological database provided in the RAI response dated 11/11/2003 and the Apr 1971 - Mar 1975 onsite data provided in FSAR Table 2.3-8B is provided below:



This wind rose shows considerable discrepancies for the following wind direction sectors:

Period of Record	Wind Direction Sector		
	SSE	S	SSW
Apr 1971 - Mar 1975	5.6%	6.5%	6.5%
Jan 2000 - Dec 2002	1.4%	1.4%	2.3%

Please explain what might have caused these discrepancies in reported wind direction frequency distributions between the 1971 - 1975 data set and the 2000 - 2002 data set.

SNC Response:

National Oceanic and Atmospheric Administration (NOAA) data for the Dothan airport and Fort Rucker were reviewed and indicate regional 4%-6% southerly wind direction frequencies for the 1999-2003 period. It has been suggested that the cooling towers, which were not operating in 1971-1975 and would not impact the NOAA data, may interrupt local on-site flow from the southerly directions; though no atmospheric studies have been performed to confirm this hypothesis. However, the current on-site data are reviewed on a continuing basis for internal consistency between the primary tower data and backup tower data. A year-to-year comparison of the 2000-2003 data set indicates no internal inconsistency or abnormal trends.

To estimate the potential impact of the direction frequency distribution differences between the 1971-1975 data and the 2000-2003 data, an additional file was constructed by copying as-recorded 2000-2003 data from the SSE, S and SSW directions into a one month data set. These data were repeated six times and added to the 2000-2003 data yielding an overall frequency from these directions of about 5% and the most limiting

Unit 1 (the unit south of the control room intakes) χ/Q was recalculated. This study indicated a small increase in the Unit 1 reactor to control room χ/Q, but the Unit 2 values remained limiting. Since common calculations with the more limiting χ/Qs are prepared for both units, the differences in frequency distributions will have no impact on results.

Atmospheric Dispersion Factors

NRC Question 14

The dose calculations assume automatic control room isolation with manual actuation of the control room emergency filtration system occurs within 10 minutes of the accident. During the isolation mode, unfiltered inleakage into the control room is assumed to be 10 cfm. This inleakage of unfiltered air, which can occur through the doorways, envelope penetrations, and ventilation system components, was apparently modeled using the control room intake χ/Q values. Please verify that there are no other potential unfiltered inleakage pathways that could result in χ/Q values that are higher than the control room intake χ/Q values.

SNC Response:

Assumed inleakage flow rates and timing are addressed above in the response to question 6. The release of activity into the containment is expected to result in rapid isolation of the containment HVAC systems. Although a small purge volume is released via the containment purge prior to isolation of the containment, leakage from the ducting into the auxiliary building will make an insignificant source concentration contribution to control room inleakage compared to the concentration modeled using the control room χ/Q. There are no other normally open penetrations from the containment air into the auxiliary building. With the hatch open, containment pressure will be nearly atmospheric; thus leakage directly from the containment into the auxiliary building is highly unlikely.

The control room envelope is surrounded by areas served by multiple non-safety related HVAC systems and the unventilated mechanical equipment room (MER) above the control room. Seven of these HVAC systems have outside air intakes located on the Auxiliary Building roof as shown in Figure 1; however, the most likely source of inleakage is the negative pressure duct work in the mechanical equipment room. As noted in the χ/Q calculation, the straight-line distance from the equipment hatch through the containment structure to the control room emergency intakes was used rather than the actual distance on the travel path around the containment. This results in a conservatively short distance that bounds the distance to other potential receptors in the direction of the control room which may be nearer the source than the emergency intakes (i.e.; the control room normal air intake on the east side of the MER or the north and south doors to the MER above the control room, Unit 1 and 2 electrical chase intakes, Unit 1 and 2 rad area intakes) except for the TSC intake and Unit 1 and 2 non-rad intakes. With the TSC intake in operation, inleakage through the TSC would be filtered and would not be a concern compared to unfiltered outside air. Thus, only the χ/Q for inleakage from the Unit 1 and 2 non-rad area might not be bounded by the control room value.

The non-rad area intakes are about 30 ° off the wind direction to the control room, so the plume centerline cannot envelope the control room intakes (and mechanical equipment room) and the non-rad area intake. Intake through the non-rad area intake would be mixed, diluted, and held up with the pre-accident, non-contaminated air in the non-rad area prior to leakage into the control room. Inleakage through the non-rad area intake is one of eight potential pathways. The χ/Q for the non-rad area intake is bounded by that for the TSC, and thus would be less than a factor of two greater than the control room χ/Q. Finally, the calculated dose results with the control room χ/Q indicate a large margin to the acceptance criteria. Therefore, using the control room χ/Q to model all of the inleakage pathways is reasonable.

NRC Question 15

Staff review indicates apparent deficiencies in the ARCON96 run inputs provided in your Request for Additional Information (RAI) response dated November 11, 2003, as discussed below. Please consider the impact of these modeling deficiencies, along with the meteorological database abnormalities discussed previously, on the ARCON96 results. If necessary, provide a set of revised χ/Q values, along with a copy of the associated ARCON96 printouts.

a) The "distance to receptor" ARCON96 input parameter should represent the horizontal distance between the release point and the receptor. However, it appears from Table 1 in Enclosure 2 to the RAI response that the "distance to receptor" values provided as input to the ARCON96 runs also include the difference in height between the release point (e.g., the containment hatch door) and the receptor (e.g., CR air intake). Note that ARCON96 uses the difference in the "release height" and "intake height" input parameters (as well as the "elevation difference" input parameter) to determine the slant path distance between the release point and the receptor for ground level releases. Since the "release height" and "intake height" input parameters for the ARCON96 runs also account for the difference in elevation between the release point and the receptor, the difference in elevation between the release point and the receptor appears to have been double-counted.

SNC Response:

The above mistake has been corrected in Version 1 of the calculation.

b) A value of 4.0 was used for the "averaging sector width constant" input parameter instead of a value of 4.3 as suggested in Table A-2 of Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants."

SNC Response:

Version 0 of the calculation was performed prior to the issuance of RG 1.194. A value of 4.3 was used for the "averaging sector width constant" input parameter in Version 1 of the calculation.

NRC Question 16

In addition to describing the inputs, assumptions, and bases used to execute ARCON96 for containment hatch door releases to the control room air intakes, the RAI response dated 11/11/2003 also provides additional information regarding vent and containment leakage releases and Technical Support Center air intake χ/Q values. ARCON96 outputs for these additional source/receptor combinations are also provided. Since the resulting χ/Q values for these additional source/receptor combinations are neither presented or discussed in the original

Containment Equipment Hatch License Amendment Request, what is the intent in providing this additional information in the RAI response submittal?

SNC Response:

A single calculation was prepared for source/receptor combinations anticipated to be needed for future analyses including post-LOCA leakage from the (reactor) containment. The vent releases were calculated to verify that other release pathways (e.g., post-LOCA recirculation loop leakage outside containment releases via the penetration room filtration system) were not more limiting than the containment leakage. The RAI response presented excerpts from the calculation including this additional information from the calculation as an efficient means of response.

Health Physics

NRC Question 17

In the event of an in-containment FHA, a designated crew of workers will be assigned to manually close the containment structure equipment hatch shield doors. A "best estimate" thyroid dose of about 46 rem was provided, based on a one-hour stay-time inside containment. Provide an estimate of whole body (deep dose) for a member of the crew to ensure that these doses are consistent with exposure guidelines of 10 CFR 50.47(b)(11). List all pertinent assumptions (e.g., airborne and external source terms, stay time, respiratory protection factors, etc.) taken to develop the dose estimates for these emergency workers. Crew member whole body doses should include external doses from all in-containment sources including worker immersion/shine dose from noble gas and iodine airborne cloud, and contained system/component sources (e.g. filters). Doses should include in-containment exposure received traversing to and from the hatch worksite.

SNC Response:

Procedures are in place to ensure that exposures are controlled consistent with Environmental Protection Agency (EPA) Emergency Worker and Lifesaving Activity Protective Action Guides as referenced in 10 CFR 50.47(b)(11). The attached Table 2 replaces Table 2 of the original submittal dated August 29, 2003 and provides the pertinent assumptions for estimating thyroid, whole body and skin doses. Based on these parameters, the doses to a crew member inside the containment for one hour would be about 46.1 REM thyroid, 1.1 REM skin, and <0.1 REM whole body. Crew members transiting to the equipment hatch from inside the containment will include their transit time in the one hour total stay time. Crew members transiting to the equipment 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. No other significant contributions to worker dose are expected since the containment purge filter is outside the containment and the pre-access filter is typically not used during refueling, is on the opposite side of the containment and is partially shielded by the steam generator and compartment walls.

NRC Question 18

For the emergency response action of closing the equipment hatch, describe the radiation protection job planning and job-site coverage and the radiation surveys, personal protection, and dose monitoring equipment that will be provided to the crew members. Describe the initial (and continuing) radiological training that will be provided, including whether the crew workers will be qualified and trained to use respiratory protection devices or other means to limit intake of radioactive materials (e.g., use of KI to minimize radioiodine uptake of the thyroid). Describe any mockup (or actual) training or practice that will be provided to the crew members that would minimize in-containment stay-time during the accident.

SNC Response:

If any monitor alarms as a result of a fuel handling accident, Health Physics (HP) will perform the following: 1) Safely evacuate personnel, 2) Contact the Control Room and HP Supervision for additional actions, 3) Ensure the area has been properly isolated, 4) Conduct additional sampling as directed. No HP job-site coverage will be required for this activity. If a fuel handling accident occurred, HP would be locating personnel and ensuring their safe evacuation from containment.

Personnel will be required to wear an electronic dosimeter and TLD when entering an RCA in support of the equipment hatch closure. Assigned equipment hatch personnel will be permitted to remain logged onto their RWP at all times while providing response capabilities. Protective clothing will be required in contaminated areas. Authorization will be given to allow personnel to wear protective clothing over their personal clothing. Respirators will not be required for the entries to provide emergency closure of the equipment hatch. In the event that contamination of personnel occurs, whole body counts will be utilized to assess the situation. If additional assessments are required to determine suspected or known intakes of radioactive material, follow-up bioassay sampling and analysis may be conducted.

The radiological conditions in the area of the equipment hatch will be assessed through the air sampling program.

- A Continuous Air Monitor will be in service inside containment on the 155' elevation and the area of the equipment hatch outside of containment.
- Continuous air sampling will be required in containment anytime the equipment hatch is open and fuel movement is in progress.
- Continuous air sampling will be required at the equipment hatch any time the equipment hatch is open and fuel movement is in progress.
- Air sampling will be for particulate and iodine activity. If the situation warrants, noble gas samples will also be taken.

Many Maintenance personnel are trained to use respiratory devices or other methods to limit the intake of radioactive material through General Employee Training. There will be no additional training required to support closure activities. However, as noted above, respirators will not be required for the entries to provide emergency closure of the equipment hatch.

Although not anticipated, if a person were to be exposed to airborne radioactive iodine such that they would exceed 2000 Derived Air Concentration (DAC)-hrs, the issuance of potassium iodide as a thyroid blocking agent would be considered. Procedures are in place to provide guidance for this activity should the need arise.

Maintenance personnel have procedural guidance for normal closure of the equipment hatch and perform the activity routinely during refueling outages. Training and actual performance or simulation of equipment hatch closure is required of Mechanical Maintenance Journeymen.

Table 2 (sheet 1 of 2)

REALISTIC PARAMETERS USED IN FUEL HANDLING ACCIDENT ANALYSIS
(For Operators Designated to Close the Equipment Hatch)

Core thermal power (MWt)	2775
Time between plant shutdown and accident (hr)	120
Minimum water depth between tops of Damaged fuel rods and water surface (ft)	23
Damage to fuel assembly	All rods in one row (17) ruptured
Activity release from assembly	Calculated iodine activity per FSAR Table 15.1-4 Noble gas activity in ruptured rods per RG 1.195, Table 2
Radial peaking factor	1.0
Decontamination factor in water	
Elemental iodine (99.75%)	500
Organic iodine (0.25%)	1
Noble gases	1
Release to containment (Ci)	
I^{131}	1.9
Kr^{85}	2.76×10^{1}
Xe^{131m}	1.21×10^{1}
Xe^{133m}	2.05×10^{1}
Xe^{133}	1.49×10^{3}
Mixing volume in containment ft^3 (1)	2×10^{6}
Exposure volume (ft^3)(2)	1.64×10^{6}
Isolation time	N/A
Iodine filtration system	Containment pre-access or purge system (not credited)
Operator breathing rate (m^3/s)(3)	3.47×10^{-4}
Exhaust flow rate (cfm) (4)	53500

Table 2 (sheet 2 of 2)

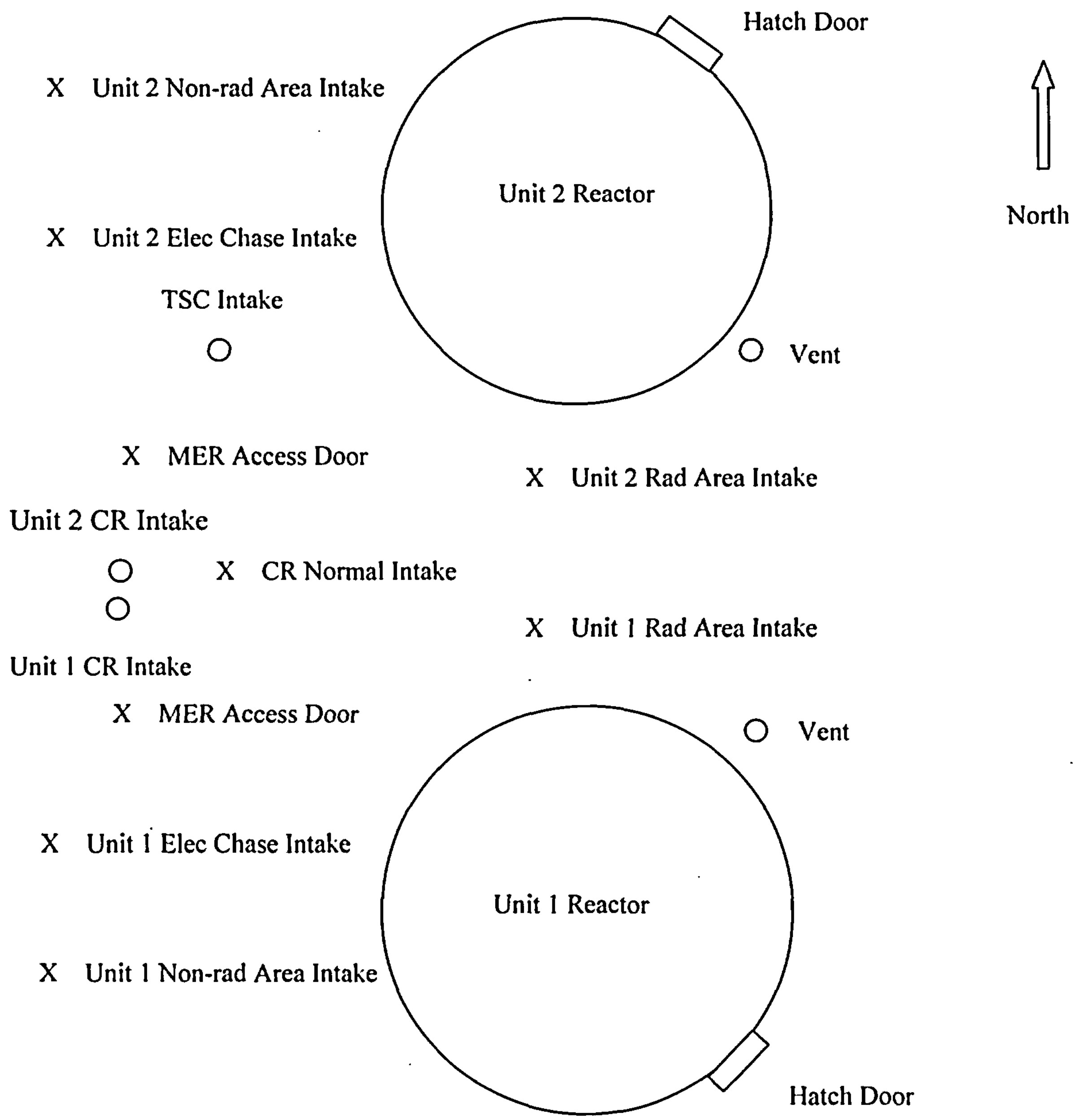
Atmospheric dilution estimates (s/m^3)[(5)]	
0-2 h	1.81×10^{-3}
Dose conversion factors	ICRP 30

Notes

(1) 100% mixing of the volume by containment fan coolers.
(2) The exposure volume is the free volume above the operating deck, used to compute the Geometry Factor for a finite cloud volume (GF = 9.3).
(3) No credit for respiratory protection.
(4) Used for exposure during travel to the equipment hatch from outside containment only. Radiation monitors are expected to alarm and isolate the containment. Doses inside containment do not credit reduction of containment inventory by the exhaust or filtration systems.
(5) Most limiting equipment hatch-TSC intake values recalculated with ARCON96, used for exposure during travel to the equipment hatch from outside containment.

Figure 1

Air Intake Locations and Release Points



Note: Release point from the reactor was a point on the containment surface closest to the receptor location.

ENCLOSURE 2

Joseph M. Farley Nuclear Plant Units 1 and 2
Response to Second Request for Additional Information Related to
Request to Revise Technical Specifications - Containment Equipment Hatch

Excerpt from Design Bases Fuel Handling Calculation

Purpose : The purpose of this calculation is to prepare offsite (EAB and LPZ) and control room doses for a postulated fuel handling accident (FHA) in the auxiliary building, or in the containment with the equipment hatch open. The reactor is assumed to be operating at uprated conditions on an 18 month cycle as described in the body of the calculations.

Evaluations of additional cases are provided in the attachments.

Attachment 4 addresses an FHA in the containment with the doors closed, taking credit for the purge filter. The model description serves as a reference for Attachment 7, and the results have been superceded by those of Attachment 7.

Attachment 5 addresses an FHA in the Auxiliary Building with 0.5% filter bypass flow.

Attachment 6 addresses an FHA in the Auxiliary Building with the hatches in the roof of the spent fuel pool area open to transfer new fuel into the spent fuel pool. This model assumes no credit for the PRF filters (*i.e.*, 100% bypass).

Attachment 7 addresses an FHA in the containment with credit for the purge radiation monitor initiating isolation.

Criteria :

For case in the containment with the equipment hatch open, input assumptions will be consistent with Regulatory Guide 1.195 and 1.52, and offsite doses shall meet the criteria of NUREG-0800 as shown below. The control room operator dose criteria from RG 1.195 are 5 REM whole body, 50 REM thyroid, and 50 REM skin (may be 75 REM unprotected if the licensee commits to protective eye wear and clothing).

For the Auxiliary Building cases or containment case with the equipment hatch closed in the Attachments, input assumptions will be consistent with Regulatory Guides (RG) 1.25 and 1.52, with the exception of the fraction of I_{131} in the clad gap, which will be consistent with NUREG/CR-5009 for high burnup fuel, and χ/Q's which will be consistent with the current (Rev. 17) FSAR values. The results of the calculations are acceptable if the offsite doses meet the criteria of NUREG-0800 (Standard Review Plan, SRP) and NUREG-75/034 (FNP SER and supplements) and the control room dose meets 10 CFR 50, Appendix A, General Design Criterion 19 (GDC 19):

	Thyroid Dose (Rem)	Whole Body Dose (Rem)
Site Boundary	75	6.3
LPZ Boundary	75	6.3
Control Room	30	5

Changes to the results of Attachment 5, evaluated in Attachment 6, must also meet the "minimal increase" criteria of 10 CFR 50.59 and reference 24. (See Attachment 6 for additional details.)

Conclusions :

The fuel handling accident in the containment with the equipment hatch open, analyzed in accordance with RG 1.195 results in offsite doses which meet the acceptance criteria:

	Thyroid dose (Rem)	Skin dose (Rem)	Whole body dose (Rem)
Site Boundary	68.2	0.4	0.2
Low Population Zone	25.1	0.2	0.1
Control Room	5.6	0.3	<0.1

The realistic estimate of the thyroid dose to the individual(s) designated to close the equipment hatch, with no credit for a respirator or other protective gear is 23.4 Rem, which meets the RG 1.195 limit for operators. The maximum exposure time to remain within the 50 Rem limit would be about 65 minutes.

From Attachment 5, the FHA in the auxiliary building meets the acceptance criteria assuming credit for the operation of the penetration area filtration system with 0.5% bypass flow.

	Thyroid dose (Rem)	Skin dose (Rem)	Whole body dose (Rem)
Site Boundary	21.6	0.8	0.4
Low Population Zone	7.9	0.3	0.1
Control Room	10.3	-	-

From Attachment 6, the FHA in the auxiliary building without PRF filters meets the "minimal increase" acceptance criteria IF the most recently discharged fuel in the spent fuel pool has decayed at least 676 hours since discharge from the reactor.

	Site Boundary	Low Population Zone	Control Room
Thyroid Dose (Rem)	25.7	9.5	12.3

From Attachment 7, the FHA in the containment with the doors closed and no credit for the purge filter meets the acceptance criteria:

	Site Boundary	Low Population Zone	Control Room
Thyroid Dose (Rem)	12.1	4.5	28.6
Whole Body Dose (Rem)	0.4	0.1	0.8

Major Equations : The calculations are prepared using the TACT V computer code, installed on a NEC P90 computer. Proper operation of the program was verified by running the test problems and comparing the output to that provided in reference 10. A listing of the TACT 5 directory is included in the Attachment 1. The equations for the numerical solutions for activity transport, and the equations for the resultant doses are described and discussed in reference 10. Equations used to derive input values for the computer analyses are explained in the body of the calculation.

Assumptions : Offsite dose analyses are prepared using dose conversion factors from ICRP 30 as provided in the users manual for TACT5 (reference 10). This is consistent with the NRC SERs for similar recent analyses (references 13 and 14).

Assumptions for the FHA in the auxiliary building or containment with the equipment hatch closed are consistent with the guidelines of Regulatory Guide 1.25, except as noted above, and are shown in Table 1. It is assumed that the air space above the spent fuel pool is maintained at a negative pressure relative to adjacent areas so that radioactive releases are processed through the penetration area filtration system--this implies that the doors and hatches into the area are closed or allow flow only into the spent fuel handling area, thereby preventing bypass of the filtration system.

Assumptions for the FHA in the containment with the equipment hatch open are consistent with the applicable portions of RG 1.195. The major differences are in the activity transport assumptions, which are dependent on the functioning of the various HVAC systems in the containment and the volume served.

References :

1. USNRC Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Reactors," Revision

2. USNRC Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision

3. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988.

4. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 15.7.4, Revision 1, July 1981.

5. NUREG 75/034, "Safety Evaluation Report ... Joseph M. Farley Nuclear Plant Units 1 and 2," May 2, 1975 and supplements.

6. 10 CFR 50, Appendix A, General Design Criterion 19, "Control Room."

7. FNP Drawings:

a. D175015, Revision 7	
b. D175022, Revision 18	Fuel Pool Area P & IDs
c. D175045, Revision 13	
d. D175092, Revision 6	
e. D175103, Revision 11	Containment duct layout
f. D175104, Revision 3	
g. D176069, Revision 15	Building volumes
h. D175067, Revision 7	
i. D176072, Revision 3	
j. D175028, Revision 12	Containment HVAC flow rates
k. U359090	
l. D205012, Revision 39	Control Room HVAC P & ID

8. FNP Uprate Analysis Input Assumptions, Westinghouse Letter ALA-95-756, 12/15/95.

9. Final Core Inventory Source Terms, Westinghouse Letter ALA-96-508, 2/1/96.

10. TACT5 Documentation
    a. NUREG/CR-5106, "User's Guide for the TACT5 Computer Code," June 1988.
    b. NS-94-02, "Verification of TACT5," 4/15/92.

11. FNP FSAR
    a. Section 15.4.5 and Table 15.4-27 and 29
    b. Table 2.3-12
    c. Section 9.3.1
    d. Sections 3K and 6.5
    e. Table 15.1-4

12. Westinghouse Letter ALA-94-797, "Revision of Fuel Handling Offsite Doses," 12/29/94.

13. STP Units 1 and 2 - Amendment Nos. 69 and 58 to Facility Operating License NPF-76 and NPF-80, dated February 9, 1995.

14. Prarie Island Nuclear Generating Plant, Unit Nos. 1 and 2 - Issuance of Amendments Re: Control Of Airlock Doors During Core Alterations, Dated July 3, 1995.

15. FNP-1-RCP-252, "Radiation Monitoring System Setpoints," Revision 30.

16. FNP Calculation 38.7

17. DCP 95-0-8863-1-1 (Control room normal HVAC)

18. FNP Calculations 42.01, Rev. 0 and 41.07, Rev. 0.

19. FNP Technical Specification 3/4.7.7

20. Victoreen Radiation Monitoring Tech. Manual, U-262532, Rev. J.

21. DCPs 95-1-8887, 8888, and 8889 (Air Compressor replacements)

22. ANSI/ANS 58.8-1994, "Time Response Design Criteria for Safety-related Operator Action".

23. Letters to the NRC, LCV-0527, "VOGTLE ELECTRIC GENERATING PLANT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ALLOW CONTAINMENT AIRLOCK DOORS TO BE OPEN CURING FUEL MOVEMENT", dated March 17, 1995; and LCV-1149-D, VOGTLE ELECTRIC GENERATING PLANT REQUEST TO REVISE TECHNICAL SPECIFICATIONS CONTAINMENT EQUIPMENT HATCH.

24. NEI 96-07, Rev. 1, "Guidelines for 10 CFR 50.59 Implementation," as endorsed by Regulatory Guide 1.187, Rev. 0, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments."

25. USNRC Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," 5/03.

26. Calculation BM-03-0018-001, "Control Room and Technical Support Center Air Intake X/Q Estimates," 6/13/03.

27. Technical Specifications, Section 5.5.11

28. "Fuel Handling Accident (FHA) Drop Height Reference Information," Westinghouse Letter ALA-01-057, June 12, 2001.

Body of Calculation :

A. Source Terms

Source terms for the FHA are developed from Westinghouse input of core activities for the uprated power of 2775 MWt (reference 8), plus 2% for calorimetric errors (2831 MWt, reference 9). Parametric analyses were performed by Westinghouse to determine how these source terms might vary with different fuel designs (enrichment, blankets, burnup) to define a "bounding" factor to be applied to the base core inventory. Because the resultant thyroid, whole body and skin doses are determined by iodines and noble gases, the parametric analyses for these isotopes were examined and the base core inventory is multiplied by 1.02 to bound the FHA source terms.

For the FHA in the containment with the equipment hatch open, the gas gap inventory is 5 % of the core inventory except Kr_{85} is 10% and I_{131} is 8 % consistent with reference 25. The TACT V code uses Ci/MWt as input, so the core inventory is then divided by 2775 and input without decay to file FNPGAP30:

$$\text{Inventory} \times 1.02 \times 0.05\ (\text{ex } Kr_{85} = 0.1 \text{ and } I_{131} = 0.08) \div 2775 = \text{Ci/MWt.}$$

The remaining data in FNPGAP31 is from file MLWRICRP.30 from reference 10a, Appendix E

```
   2      3      3      0      0      1      1
WHOLEBDY  SKIN  THYROID
HALOGENS NOBLES
  ELEM.   ORG.    PART.
  I  131
 9.963996E-07 2.219000E+03
 5.590000E-02 3.070000E-02 1.100000E 06
 0.0          0.0          0.0
 0.0          0.0          0.0
   1      1      0      0      0      0      0
  I  132
 8.269001E-05 2.011000E+03
 3.550000E-01 1.100000E-01 6.300000E 03
 0.0          0.0          0.0
 0.0          0.0          0.0
   2      1      0      0      0      0      0
  I  133
 9.219000E-06 2.858000E+03
 9.110000E-02 8.900000E-02 1.800000E 05
 0.0          0.0          0.0
 0.0          0.0          0.0
   3      1      0      0      0      0      0
  I  134
 2.228000E-04 3.143000E+03
 4.110000E-01 1.420000E-01 1.100000E 03
 0.0          0.0          0.0
 0.0          0.0          0.0
   4      1      0      0      0      0      0
 I  135
2.864000E-05 2.672000E+03
2.490000E-01 7.860000E-02 3.100000E 04
0.0          0.0          0.0
0.0          0.0          0.0
  5      1      0      0      0      0      0
 I  136
8.349001E-03 1.266000E+03
6.786000E-01 1.300000E+00 0.0
0.0          0.0          0.0
0.0          0.0          0.0
  6      1      0      0      0      0      0
 KR  83M
1.035000E-04 1.790000E+02
1.270000E-05 0.000000E+00 0.0
0.0          0.0          0.0
0.0          0.0          0.0
  7      2      0      0      0      0      0
 KR  85M
4.385000E-05 3.827000E+02
2.310000E-02 4.970000E-02 0.0
0.0          0.0          0.0
0.0          0.0          0.0
  8      2      0      0      0      0      0
 KR  85
2.042000E-09 2.657000E+01
3.310000E-04 4.840000E-02 0.0
0.0          0.0          0.0
0.0          0.0          0.0
  9      2      0      0      0      0      0
```

```
 KR  87
1.519000E-04 7.380000E+02
1.330000E-01 3.360000E-01 0.0
0.0          0.0          0.0
0.0          0.0          0.0
 10     2     0     0     0     0     0
 KR  88
6.875000E-05 1.040000E+03
3.380000E-01 7.760000E-02 0.0
0.0          0.0          0.0
0.0          0.0          0.0
 11     2     0     0     0     0     0
 XE 131M
6.680000E-07 1.546000E+01
1.250000E-03 1.330000E-02 0.0
0.0          0.0          0.0
0.0          0.0          0.0
 12     2     0     0     0     0     0
 XE 133M
3.490000E-06 8.900000E+01
4.290000E-03 2.960000E-02 0.0
0.0          0.0          0.0
0.0          0.0          0.0
 13     2     0     0     0     0     0
```

```
 XE 133
1.522000E-06 2.755000E+03
4.960000E-03 9.670000E-03 0.0
0.0          0.0          0.0
0.0          0.0          0.0
 14     2     0     0     0     0     0
 XE 135M
7.400000E-04 5.550000E+02
6.370000E-02 2.140000E-02 0.0
0.0          0.0          0.0
0.0          0.0          0.0
 15     2     0     0     0     0     0
 XE 135
2.091999E-05 6.390000E+02
3.590000E-02 6.320000E-02 0.0
0.0          0.0          0.0
0.0          0.0          0.0
 16     2     0     0     0     0     0
 XE 137
2.961000E-03 2.498000E+03
2.830000E-02 4.590000E-01 0.0
0.0          0.0          0.0
0.0          0.0          0.0
 17     2     0     0     0     0     0
 XE 138
6.796001E-04 2.378000E+03
1.870000E-01 1.470000E-01 0.0
0.0          0.0          0.0
0.0          0.0          0.0
 18     2     0     0     0     0     0
```

C. Containment Building Model: Regulatory Guide 1.195 FHA in Containment (with Equipment Hatch Open)

For an FHA in the containment, the source terms are modeled as above in section A. The building space used for the dilution of the activity released is evaluated for the space served by the containment purge system. During refueling, the containment may be purged using the main purge, drawing air from elevations 130 and 155 of the containment (references 7d, e, f). [NOTE that the pool sweep is typically not used, and has been disconnected from the purge exhaust system. If it were to be used, it would exhaust to elevation 139 below the operating deck.] Assuming mixing above the pool up to the elevation of the containment cooling fan header (but no mixing outside this envelope which might be induced by operation of the fans), and 90% free space, this represents a volume of approximately:

$$\pi \times 65^2 \times 55 \times 0.9 \approx 6.6 \times 10^5 \text{ ft}^3.$$

An FHA in the refueling canal will release the gap activity from one assembly (reference 11, 12) with the highest power, *i.e.* with a peaking factor of 1.7 (reference 8, which exceeds the reference 1 value), to the pool water which will scrub the iodines as they evolve into the building air. Release factors from reference 25 are 8% I^{131}, 10% Kr^{85}, and 5% for other iodines and noble gases. The original partition between elemental and organic species of iodine and the cleanup by the pool water as described in reference 25 are combined to result in 0.5% of the iodine instantaneously released to the building, partitioned as 50% organic and 50% elemental.

Releases are driven from the containment by the purge system flow. Although the initial release concentration in the containment:

$$797.8 \text{ Ci } Kr^{85} \times 1 \text{ E6 } \mu\text{Ci/Ci} \div 6.6 \text{ E5 ft}^3 \div 28317 \text{ cc/ft}^3 = 4 \times 10^{-2} \mu\text{Ci/cc}$$

is well in excess of the containment purge isolation radiation monitor setpoint of 1×10^{-4} μCi/cc, no credit for stopping the purge flow or purge filter is modeled. Then the main purge flow plus 10% exhausts

$$48{,}500 \times 1.1 \text{ cfm} \times 120 \text{ min} / 6.6 \times 10^5 \text{ ft}^3 = 9.7 \text{ times the containment volume,}$$

which exceeds the requirements of reference 25. Releases are modeled as ground level, with χ/Q values taken from reference 26. The annotated input file (FHAHTCH3.in) and offsite dose results are shown below, where control room parameters are taken from section D below.

FHAHTCH3.in

```
'c:fnpgap31 '                                                   RG 1.195 source terms from
'c:LTAPE   ','c:MTAPE   ','c:NTAPE   '                                    section A above.
'    Farley FHA in the Containment with Equip Hatch Open (RG 1.195)  '
'                                                           '
 0, 0, 1, 1, 0
  2,  2
'Contnmnt','CtrlRoom'
 25
 1.7675E+1, -1.00E+02                               2775 MWt/157 assys, 100 hr decay
 8.500E-03, 1.700E-00                               1.7/400 iodine and 1.7 noble gas
 0.000E+00, 0.000E+00                                           release fractions
 5.000E-01, 5.000E-01, 0.000E+00                    50% elemental, 50% organic iodine
 1.000E+00, 0.000E+00, 0.000E+00                    100% elemental noble gases
 6.600E+05, 1.140E+05                               Containment refueling area and
'TIME INTERVAL   ',0,0,0,0,2, 0.000E+00, 2.7778E-06             control room volumes
'INITIAL FRACTION',0,0,0,0,2, 1.000E+00, 0.000E+00
'FILTER EFF      ',1,1,0,1,3, 0.000E+00, 0.000E+00, 0.000E+00
'FILTER EFF      ',1,2,0,1,3, 0.000E+00, 0.000E+00, 0.000E+00
'FILTER EFF      ',1,3,0,1,3, 0.000E+00, 0.000E+00, 0.000E+00
'TRANSFER CFM    ',0,0,0,1,3, 5.350E+04, 0.000E+00, 2.893E+01     HVAC flow rates
'TRANSFER CFM    ',0,0,0,2,3, 1.360E+03, 0.000E+00, 1.000E+00     Normal CR exhaust flow
'DOSE PARAMS     ',0,0,0,0,5, 7.600E-04, 3.470E-04, 2.800E-04, 3.470E-04, 0.000E+00
'TIME INTERVAL   ',0,0,0,0,2, 2.778E-06, 2.778E-04
'TIME INTERVAL   ',0,0,0,0,2, 2.778E-04, 8.333E-03
'TIME INTERVAL   ',0,0,0,0,2, 8.333E-03, 1.250E-02
'TIME INTERVAL   ',0,0,0,0,2, 1.250E-02, 2.500E-02         45 sec, CR auto-isolation
'FILTER EFF      ',1,1,0,1,3, 0.000E+00, 0.000E+00, 0.000E+00
'FILTER EFF      ',1,2,0,1,3, 0.000E+00, 0.000E+00, 0.000E+00
'FILTER EFF      ',1,3,0,1,3, 0.000E+00, 0.000E+00, 0.000E+00
'TRANSFER CFM    ',0,0,0,1,3, 5.350E+04, 0.000E+00, 4.201E+00     Unpressurized CR intake
'TRANSFER CFM    ',0,0,0,2,3, 1.975E+02, 0.000E+00, 1.000E+00          flow and exhaust
'DOSE PARAMS     ',0,0,0,0,5, 7.600E-04, 3.470E-04, 2.800E-04, 3.470E-04, 0.000E+00
'TIME INTERVAL   ',0,0,0,0,2, 2.500E-02, 4.167E-02
'TIME INTERVAL   ',0,0,0,0,2, 4.167E-02, 8.333E-02
'TIME INTERVAL   ',0,0,0,0,2, 8.333E-02, 1.667E-01
'TIME INTERVAL   ',0,0,0,0,2, 1.667E-01, 3.333E-01       10 min, manual pressurization
'FILTER EFF      ',1,1,0,1,3, 0.000E+00, 0.000E+00, 9.594E+01 Equivalent pressurization
'FILTER EFF      ',1,2,0,1,3, 0.000E+00, 0.000E+00, 9.594E+01     filter efficiency
'FILTER EFF      ',1,3,0,1,3, 0.000E+00, 0.000E+00, 0.000E+00
'FILTER EFF      ',1,1,0,2,3, 0.000E+00, 0.000E+00, 9.450E+01  Recirculation filter
'FILTER EFF      ',1,2,0,2,3, 0.000E+00, 0.000E+00, 9.450E+01     efficiency
'FILTER EFF      ',1,3,0,2,3, 0.000E+00, 0.000E+00, 0.000E+00
'TRANSFER CFM    ',0,0,0,1,3, 5.350E+04, 0.000E+00, 8.188E+00     Pressurized CR intake
'TRANSFER CFM    ',0,0,0,2,3, 3.850E+02, 0.000E+00, 2.700E+03          flow and exhaust
'DOSE PARAMS     ',0,0,0,0,5, 7.600E-04, 3.470E-04, 2.800E-04, 3.470E-04, 0.000E+00
'TIME INTERVAL   ',0,0,0,0,2, 3.333E-01, 6.666E-01
'TIME INTERVAL   ',0,0,0,0,2, 6.666E-01, 7.500E-01
'TIME INTERVAL   ',0,0,0,0,2, 7.500E-01, 1.000E+00
'TIME INTERVAL   ',0,0,0,0,2, 1.000E+00, 1.250E+00
'TIME INTERVAL   ',0,0,0,0,2, 1.250E+00, 1.500E+00
'TIME INTERVAL   ',0,0,0,0,2, 1.500E+00, 1.750E+00
```

```
'TIME INTERVAL    ',0,0,0,0,2, 1.750E+00, 2.000E+00
'TIME INTERVAL    ',0,0,0,0,2, 2.000E+00, 2.250E+00
'TRANSFER CFM     ',0,0,0,1,3, 5.350E+04, 0.000E+00, 6.253E+00   Change CR X/Q and flow
'DOSE PARAMS      ',0,0,0,0,5, 2.900E-04, 3.470E-04, 1.100E-04, 3.470E-04, 0.000E+00
'TIME INTERVAL    ',0,0,0,0,2, 2.250E+00, 2.500E+00
'TIME INTERVAL    ',0,0,0,0,2, 2.500E+00, 3.000E+00
'TIME INTERVAL    ',0,0,0,0,2, 3.000E+00, 4.000E+00
'TIME INTERVAL    ',0,0,0,0,2, 4.000E+00, 5.000E+00
'TIME INTERVAL    ',0,0,0,0,2, 5.000E+00, 6.000E+00
'TIME INTERVAL    ',0,0,0,0,2, 6.000E+00, 7.000E+00
'TIME INTERVAL    ',0,0,0,0,2, 7.000E+00, 8.000E+00
'TIME INTERVAL    ',0,0,0,0,2, 8.000E+00, 24.00E+00
'TRANSFER CFM     ',0,0,0,1,3, 5.350E+04, 0.000E+00, 3.345E+00   Change CR X/Q and flow
'DOSE PARAMS      ',0,0,0,0,5, 3.300E-05, 3.470E-04, 1.000E-05, 3.470E-04, 0.000E+00
'END              ',0,0,0,0,0, 0.000E+00, 0.000E+00, 0.000E+00
```

SUMMARY OF OFF-SITE DOSES

Farley FHA in the Containment with Equip Hatch Open (RG 1.195)
CALCULATION FOR WHOLEBDY DOSE (REMS)
MULTI NODE CONTAINMENT WITH ESF

START TIME (HRS)	EXCLUSION RADIUS EACH STEP	ACCUM.	LOW POPULATION ZONE EACH STEP	ACCUM.
0.000E+00	2.630E-06	2.630E-06	9.691E-07	9.691E-07
2.778E-06	2.602E-04	2.629E-04	9.588E-05	9.685E-05
2.778E-04	7.470E-03	7.733E-03	2.752E-03	2.849E-03
8.333E-03	3.751E-03	1.148E-02	1.382E-03	4.231E-03
1.250E-02	1.081E-02	2.229E-02	3.981E-03	8.212E-03
2.500E-02	1.342E-02	3.571E-02	4.945E-03	1.316E-02
4.167E-02	2.915E-02	6.486E-02	1.074E-02	2.390E-02
8.333E-02	4.324E-02	1.081E-01	1.593E-02	3.983E-02
1.667E-01	4.798E-02	1.561E-01	1.768E-02	5.750E-02
3.333E-01	3.079E-02	1.869E-01	1.134E-02	6.885E-02
6.666E-01	2.526E-03	1.894E-01	9.307E-04	6.978E-02
7.500E-01	3.550E-03	1.930E-01	1.308E-03	7.109E-02
1.000E+00	1.051E-03	1.940E-01	3.874E-04	7.148E-02
1.250E+00	3.119E-04	1.943E-01	1.149E-04	7.159E-02
1.500E+00	9.298E-05	1.944E-01	3.425E-05	7.162E-02
1.750E+00	2.815E-05	1.944E-01	1.037E-05	7.163E-02
2.000E+00	3.408E-06	1.944E-01	1.293E-06	7.164E-02
2.250E+00	1.226E-06	1.944E-01	4.651E-07	7.164E-02
2.500E+00	9.335E-07	1.944E-01	3.541E-07	7.164E-02
3.000E+00	1.034E-06	1.944E-01	3.921E-07	7.164E-02
4.000E+00	8.124E-07	1.944E-01	3.082E-07	7.164E-02
5.000E+00	6.594E-07	1.944E-01	2.501E-07	7.164E-02
6.000E+00	5.354E-07	1.944E-01	2.031E-07	7.164E-02
7.000E+00	4.348E-07	1.944E-01	1.649E-07	7.164E-02
8.000E+00	2.061E-07	1.944E-01	6.245E-08	7.164E-02
	TOTAL	1.944E-01	TOTAL	7.164E-02

CALCULATION FOR SKIN DOSE (REMS)
MULTI NODE CONTAINMENT WITH ESF

START	EXCLUSION RADIUS		LOW POPULATION ZONE	
TIME (HRS)	EACH STEP	ACCUM.	EACH STEP	ACCUM.
0.000E+00	5.524E-06	5.524E-06	2.035E-06	2.035E-06
2.778E-06	5.465E-04	5.521E-04	2.014E-04	2.034E-04
2.778E-04	1.569E-02	1.624E-02	5.780E-03	5.983E-03
8.333E-03	7.877E-03	2.412E-02	2.902E-03	8.885E-03
1.250E-02	2.269E-02	4.681E-02	8.360E-03	1.724E-02
2.500E-02	2.819E-02	7.500E-02	1.039E-02	2.763E-02
4.167E-02	6.121E-02	1.362E-01	2.255E-02	5.018E-02
8.333E-02	9.081E-02	2.270E-01	3.346E-02	8.364E-02
1.667E-01	1.008E-01	3.278E-01	3.713E-02	1.208E-01
3.333E-01	6.467E-02	3.925E-01	2.383E-02	1.446E-01
6.666E-01	5.306E-03	3.978E-01	1.955E-03	1.465E-01
7.500E-01	7.457E-03	4.052E-01	2.747E-03	1.493E-01
1.000E+00	2.209E-03	4.074E-01	8.137E-04	1.501E-01
1.250E+00	6.553E-04	4.081E-01	2.414E-04	1.504E-01
1.500E+00	1.954E-04	4.083E-01	7.199E-05	1.504E-01
1.750E+00	5.922E-05	4.084E-01	2.182E-05	1.504E-01
2.000E+00	7.190E-06	4.084E-01	2.727E-06	1.504E-01
2.250E+00	2.604E-06	4.084E-01	9.878E-07	1.504E-01
2.500E+00	2.014E-06	4.084E-01	7.637E-07	1.504E-01
3.000E+00	2.262E-06	4.084E-01	8.582E-07	1.505E-01
4.000E+00	1.781E-06	4.084E-01	6.756E-07	1.505E-01
5.000E+00	1.446E-06	4.084E-01	5.485E-07	1.505E-01
6.000E+00	1.175E-06	4.084E-01	4.455E-07	1.505E-01
7.000E+00	9.540E-07	4.084E-01	3.618E-07	1.505E-01
8.000E+00	4.527E-07	4.084E-01	1.372E-07	1.505E-01
	TOTAL	4.084E-01	TOTAL	1.505E-01

CALCULATION FOR THYROID DOSE (REMS)
MULTI NODE CONTAINMENT WITH ESF

START TIME (HRS)	EXCLUSION RADIUS EACH STEP	EXCLUSION RADIUS ACCUM.	LOW POPULATION ZONE EACH STEP	LOW POPULATION ZONE ACCUM.
0.000E+00	9.227E-04	9.227E-04	3.399E-04	3.399E-04
2.778E-06	9.129E-02	9.222E-02	3.363E-02	3.397E-02
2.778E-04	2.620E+00	2.713E+00	9.654E-01	9.994E-01
8.333E-03	1.316E+00	4.028E+00	4.847E-01	1.484E+00
1.250E-02	3.790E+00	7.819E+00	1.396E+00	2.881E+00
2.500E-02	4.709E+00	1.253E+01	1.735E+00	4.616E+00
4.167E-02	1.023E+01	2.275E+01	3.767E+00	8.383E+00
8.333E-02	1.517E+01	3.793E+01	5.590E+00	1.397E+01
1.667E-01	1.684E+01	5.477E+01	6.204E+00	2.018E+01
3.333E-01	1.081E+01	6.558E+01	3.983E+00	2.416E+01
6.666E-01	8.872E-01	6.646E+01	3.269E-01	2.449E+01
7.500E-01	1.247E+00	6.771E+01	4.594E-01	2.495E+01
1.000E+00	3.692E-01	6.808E+01	1.360E-01	2.508E+01
1.250E+00	1.094E-01	6.819E+01	4.029E-02	2.512E+01
1.500E+00	3.239E-02	6.822E+01	1.193E-02	2.513E+01
1.750E+00	9.602E-03	6.823E+01	3.538E-03	2.514E+01
2.000E+00	1.088E-03	6.823E+01	4.125E-04	2.514E+01
2.250E+00	3.239E-04	6.823E+01	1.228E-04	2.514E+01
2.500E+00	1.267E-04	6.823E+01	4.806E-05	2.514E+01
3.000E+00	1.420E-05	6.823E+01	5.385E-06	2.514E+01
4.000E+00	6.126E-07	6.823E+01	2.323E-07	2.514E+01
5.000E+00	1.116E-07	6.823E+01	4.234E-08	2.514E+01
6.000E+00	2.356E-08	6.823E+01	8.938E-09	2.514E+01
7.000E+00	5.004E-09	6.823E+01	1.898E-09	2.514E+01
8.000E+00	1.535E-10	6.823E+01	4.652E-11	2.514E+01
	TOTAL	6.823E+01	TOTAL	2.514E+01

D. Control Room Model

The control room normally operates with an air supply of 1350 cfm (ref. 17) coming from the computer room A/C. Approximately 35% of this is outside air; however for conservatism, the total air flow is assumed to be outside air. TACT5 will not allow transfer from the environment back into a node and allows only one transfer path for both filtered intake and unfiltered inleakage, so an equivalent direct transfer from the containment via the environment to the control room is modeled as:

Release Rate x χ/Q x Unit Conversion Factors x Intake Rate.

The effective filter efficiency is modeled as:

1 - [(1- efficiency) x Filtered Flow + Unfiltered Flow] / [Filtered Flow +Unfiltered Flow].

The control room intake radiation monitor will isolate the control room HVAC, but manual action is required to start the pressurization and recirculation mode of operation. The pressurization and recirculation flows are assumed to be the Technical Specification (ref. 27) values of 300 +25%-10%, and 3000 ± 10%. The control room filter efficiency modeled includes a reduction of 0.5% allowed for bypass testing yielding:

CR volume = 114,000 ft^3 (ref. 16)
Pressurization flow = 375 cfm, efficiency = 98.5%
Recirculation flow = 2700 cfm, efficiency = 89.5%
Normal flow = 1350 cfm, efficiency = 0%
Unfiltered inleakage = 10 cfm, efficiency = 0% (assumed)

For the period between automatic isolation and manual pressurization, the control room unfiltered inleakage is assumed to be ½ the pressurization flow plus unfiltered inleakage = 187.5 + 10 = 197.5 cfm. Then for a containment purge flow rate of 53,500 cfm, the direct transfer rates and filter efficiencies are:

Release flow (cfm)	χ/Q (s/m^3)	Unit Conversion Factors (min/s)	Unit Conversion Factors (m^3/ft^3)	Intake flow (cfm)	Direct transfer (cfm)	Effective Filter Efficiency
53,500	8.42E-4	1/60	1/35.3	1360	28.93	0
↓	↓	↓	↓	197.5	4.201	0
↓	↓	↓	↓	385	8.188	1 - [375 (1- 0.985) +10]/(375 + 10) = 95.94%
↓	6.43E-4	↓	↓	↓	6.253	↓
↓	3.44E-4	↓	↓	↓	3.345	↓

The switch from normal HVAC intake (1350 cfm) to unpressurized unfiltered inleakage (187.5 cfm) occurs when the intake radiation monitor responds to the released activity. The 1E-4 μCi/ml monitor setpoint (ref. 15) is exceeded within about 1 second of the activity reaching the monitor:

Released activity = 1.08E6 μCi χ/Q = 8.42E-4 (s/m^3)

(1.08E6 μCi / 1 sec) x (8.42E-4 s/m^3) / (1E6 ml/m^3) ≈ 9 E-4 μCi/ml

Based on a review of reference 20, this should result in an isolation signal within about 15-20 seconds, so use 45 seconds to conservatively bound the AOV (HV3622-3629) stroke time.

The resultant control room activities are entered into an EXCEL spreadsheet to calculate the control room thyroid, skin and whole body doses. Thyroid doses are calculated for each time step from

(Average Ci-hr) x (3600 s/hr) x (35.3 ft^3/m^3) x (3.47E-4 m^3 inhaled/s) x DCF (REM/Ci)/114,000 ft^3

and whole body and skin doses are calculated from

(1/Geometry factor) x (Average Ci-hr) x (35.3 ft^3/m^3) x DCF (REM-m^3/Ci-hr)/114,000 ft^3

where the geometry factor is applied only to the whole body dose and is $1173/\ 114000^{0.338} = 22.91$ (ref 25).

FHAHTCH3.out Control Room (Curies) Inventory

ISOTOPE	Time (hr) 2.78E-06	8.33E-03	1.25E-02	2.50E-02	4.17E-02	8.33E-02
I 131	8.51E-07	2.49E-03	3.70E-03	4.20E-03	4.82E-03	6.17E-03
I 131	8.51E-07	2.49E-03	3.70E-03	4.20E-03	4.82E-03	6.17E-03
I 132	1.30E-19	3.81E-16	5.64E-16	6.38E-16	7.29E-16	9.21E-16
I 132	1.30E-19	3.81E-16	5.64E-16	6.38E-16	7.29E-16	9.21E-16
I 133	5.68E-08	1.66E-04	2.47E-04	2.80E-04	3.21E-04	4.11E-04
I 133	5.68E-08	1.66E-04	2.47E-04	2.80E-04	3.21E-04	4.11E-04
I 134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I 134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I 135	4.88E-11	1.43E-07	2.12E-07	2.40E-07	2.76E-07	3.51E-07
I 135	4.88E-11	1.43E-07	2.12E-07	2.40E-07	2.76E-07	3.51E-07
KR 83M	2.59E-21	7.55E-18	1.12E-17	1.26E-17	1.44E-17	1.82E-17
KR 85M	1.17E-11	3.43E-08	5.08E-08	5.76E-08	6.59E-08	8.38E-08
KR 85	5.83E-06	1.71E-02	2.53E-02	2.88E-02	3.30E-02	4.23E-02
KR 87	2.89E-28	8.43E-25	1.25E-24	1.41E-24	1.60E-24	2.00E-24
KR 88	4.07E-15	1.19E-11	1.76E-11	2.00E-11	2.28E-11	2.89E-11
XE 131M	2.67E-06	7.82E-03	1.16E-02	1.32E-02	1.51E-02	1.93E-02
XE 133M	5.56E-06	1.63E-02	2.42E-02	2.75E-02	3.15E-02	4.03E-02
XE 133	3.50E-04	1.03E+00	1.52E+00	1.73E+00	1.98E+00	2.53E+00
XE 135	7.52E-08	2.20E-04	3.27E-04	3.71E-04	4.25E-04	5.42E-04

Control Room Inventory (Curies)

1.67E-01	3.33E-01	6.67E-01	7.50E-01	1.00E+00	1.25	1.5	1.75	2	2.25	2.5	3
8.13E-03	6.44E-03	3.93E-03	3.46E-03	2.36E-03	1.60E-03	1.09E-03	7.40E-04	5.03E-04	3.41E-04	2.32E-04	1.07E-04
8.13E-03	6.44E-03	3.93E-03	3.46E-03	2.36E-03	1.60E-03	1.09E-03	7.40E-04	5.03E-04	3.41E-04	2.32E-04	1.07E-04
1.19E-15	8.93E-16	4.94E-16	4.24E-16	2.69E-16	1.70E-16	1.07E-16	6.77E-17	4.27E-17	2.69E-17	1.70E-17	6.76E-18
1.19E-15	8.93E-16	4.94E-16	4.24E-16	2.69E-16	1.70E-16	1.07E-16	6.77E-17	4.27E-17	2.69E-17	1.70E-17	6.76E-18
5.40E-04	4.25E-04	2.57E-04	2.26E-04	1.53E-04	1.03E-04	6.96E-05	4.69E-05	3.16E-05	2.13E-05	1.44E-05	6.52E-06
5.40E-04	4.25E-04	2.57E-04	2.26E-04	1.53E-04	1.03E-04	6.96E-05	4.69E-05	3.16E-05	2.13E-05	1.44E-05	6.52E-06
0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
4.59E-07	3.57E-07	2.11E-07	1.84E-07	1.23E-07	8.13E-08	5.39E-08	3.57E-08	2.36E-08	1.57E-08	1.04E-08	4.55E-09
4.59E-07	3.57E-07	2.11E-07	1.84E-07	1.23E-07	8.13E-08	5.39E-08	3.57E-08	2.36E-08	1.57E-08	1.04E-08	4.55E-09
2.32E-17	3.27E-17	3.34E-17	3.24E-17	2.87E-17	2.50E-17	2.17E-17	1.88E-17	1.63E-17	1.41E-17	1.22E-17	9.18E-18
1.09E-07	1.59E-07	1.75E-07	1.72E-07	1.61E-07	1.48E-07	1.36E-07	1.24E-07	1.14E-07	1.04E-07	9.48E-08	7.92E-08
5.58E-02	8.35E-02	9.66E-02	9.65E-02	9.39E-02	8.99E-02	8.57E-02	8.15E-02	7.75E-02	7.37E-02	7.00E-02	6.33E-02
2.52E-24	3.45E-24	3.32E-24	3.17E-24	2.69E-24	2.25E-24	1.87E-24	1.55E-24	1.29E-24	1.07E-24	8.84E-25	6.08E-25
3.74E-11	5.37E-11	5.72E-11	5.60E-11	5.12E-11	4.61E-11	4.13E-11	3.69E-11	3.30E-11	2.95E-11	2.64E-11	2.10E-11
2.55E-02	3.82E-02	4.41E-02	4.41E-02	4.29E-02	4.11E-02	3.91E-02	3.72E-02	3.53E-02	3.36E-02	3.19E-02	2.88E-02
5.31E-02	7.93E-02	9.14E-02	9.13E-02	8.85E-02	8.45E-02	8.02E-02	7.61E-02	7.21E-02	6.83E-02	6.48E-02	5.82E-02
3.34E+00	5.00E+00	5.77E+00	5.77E+00	5.60E+00	5.36E+00	5.10E+00	4.84E+00	4.60E+00	4.37E+00	4.14E+00	3.74E+00
7.10E-04	1.05E-03	1.19E-03	1.18E-03	1.12E-03	1.06E-03	9.87E-04	9.22E-04	8.60E-04	8.02E-04	7.49E-04	6.51E-04

Control Room Inventory (Curies)

4	5	6	7	8
2.27E-05	4.82E-06	1.02E-06	2.18E-07	4.62E-08
2.27E-05	4.82E-06	1.02E-06	2.18E-07	4.62E-08
1.07E-18	1.70E-19	2.68E-20	4.25E-21	6.72E-22
1.07E-18	1.70E-19	2.68E-20	4.25E-21	6.72E-22
1.35E-06	2.77E-07	5.72E-08	1.18E-08	2.43E-09
1.35E-06	2.77E-07	5.72E-08	1.18E-08	2.43E-09
0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
8.74E-10	1.68E-10	3.23E-11	6.22E-12	1.20E-12
8.74E-10	1.68E-10	3.23E-11	6.22E-12	1.20E-12
5.16E-18	2.91E-18	1.63E-18	9.20E-19	5.17E-19
5.52E-08	3.85E-08	2.69E-08	1.87E-08	1.31E-08
5.17E-02	4.22E-02	3.45E-02	2.81E-02	2.30E-02
2.87E-25	1.36E-25	6.42E-26	3.03E-26	1.43E-26
1.34E-11	8.55E-12	5.45E-12	3.48E-12	2.22E-12
2.34E-02	1.91E-02	1.56E-02	1.27E-02	1.03E-02
4.69E-02	3.78E-02	3.05E-02	2.46E-02	1.98E-02
3.03E+00	2.46E+00	2.00E+00	1.63E+00	1.32E+00
4.93E-04	3.74E-04	2.83E-04	2.14E-04	1.62E-04

Exposure (Ci - hr)

Time (hr)	2.78E-06	8.33E-03	1.25E-02
I 131	2.36E-12	2.08E-05	2.58E-05
I 132	3.62E-25	3.17E-18	3.93E-18
I 133	1.58E-13	1.39E-06	1.72E-06
I 134	0.00E+00	0.00E+00	0.00E+00
I 135	1.36E-16	1.19E-09	1.48E-09
KR 83M	3.59E-27	3.15E-20	3.90E-20
KR 85M	1.63E-17	1.43E-10	1.77E-10
KR 85	8.10E-12	7.12E-05	8.84E-05
KR 87	4.01E-34	3.51E-27	4.35E-27
KR 88	5.65E-21	4.96E-14	6.16E-14
XE131M	3.71E-12	3.26E-05	4.05E-05
XE133M	7.73E-12	6.79E-05	8.43E-05
XE 133	4.86E-10	4.27E-03	5.30E-03
XE 135	1.04E-13	9.18E-07	1.14E-06

Exposure (Ci - hr)

2.50E-02	4.17E-02	8.33E-02	1.67E-01	3.33E-01	6.67E-01	7.50E-01	1.00E+00	1.25E+00	1.50E+00	1.75E+00	2.00E+00
9.87E-05	1.50E-04	4.58E-04	1.19E-03	2.43E-03	3.45E-03	6.16E-04	1.45E-03	9.91E-04	6.74E-04	4.58E-04	3.11E-04
1.50E-17	2.28E-17	6.87E-17	1.76E-16	3.46E-16	4.62E-16	7.66E-17	1.73E-16	1.10E-16	6.93E-17	4.38E-17	2.76E-17
6.58E-06	1.00E-05	3.05E-05	7.92E-05	1.61E-04	2.27E-04	4.02E-05	9.46E-05	6.40E-05	4.32E-05	2.91E-05	1.96E-05
0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
5.65E-09	8.60E-09	2.61E-08	6.75E-08	1.36E-07	1.89E-07	3.29E-08	7.67E-08	5.09E-08	3.38E-08	2.24E-08	1.48E-08
1.49E-19	2.26E-19	6.79E-19	1.73E-18	4.66E-18	1.10E-17	2.74E-18	7.64E-18	6.72E-18	5.85E-18	5.07E-18	4.39E-18
6.77E-10	1.03E-09	3.12E-09	8.04E-09	2.23E-08	5.56E-08	1.45E-08	4.17E-08	3.87E-08	3.55E-08	3.25E-08	2.97E-08
3.38E-04	5.15E-04	1.57E-03	4.09E-03	1.16E-02	3.00E-02	8.05E-03	2.38E-02	2.30E-02	2.20E-02	2.09E-02	1.99E-02
1.66E-26	2.51E-26	7.50E-26	1.88E-25	4.97E-25	1.13E-24	2.71E-25	7.34E-25	6.18E-25	5.15E-25	4.28E-25	3.55E-25
2.35E-13	3.57E-13	1.08E-12	2.76E-12	7.58E-12	1.85E-11	4.72E-12	1.34E-11	1.22E-11	1.09E-11	9.78E-12	8.74E-12
1.55E-04	2.36E-04	7.18E-04	1.87E-03	5.31E-03	1.37E-02	3.68E-03	1.09E-02	1.05E-02	1.00E-02	9.53E-03	9.06E-03
3.23E-04	4.91E-04	1.50E-03	3.89E-03	1.10E-02	2.84E-02	7.62E-03	2.25E-02	2.16E-02	2.06E-02	1.95E-02	1.85E-02
2.03E-02	3.09E-02	9.41E-02	2.45E-01	6.95E-01	1.79E+00	4.81E-01	1.42E+00	1.37E+00	1.31E+00	1.24E+00	1.18E+00
4.36E-06	6.63E-06	2.01E-05	5.22E-05	1.47E-04	3.72E-04	9.85E-05	2.88E-04	2.73E-04	2.55E-04	2.39E-04	2.23E-04

Sum 0-2 hr (Ci inhaled)	Exposure (Ci - hr)								Sum 2-8 hr (Ci inhaled)
	2.25E+00	2.50E+00	3.00E+00	4.00E+00	5.00E+00	6.00E+00	7.00E+00	8.00E+00	
4.77E-06	2.11E-04	1.43E-04	1.69E-04	1.29E-04	2.75E-05	5.84E-06	1.24E-06	2.64E-07	2.66E-07
6.18E-19	1.74E-17	1.10E-17	1.19E-17	7.83E-18	1.24E-18	1.96E-19	3.11E-20	4.92E-21	1.92E-20
3.13E-07	1.32E-05	8.92E-06	1.04E-05	7.87E-06	1.62E-06	3.35E-07	6.90E-08	1.42E-08	1.64E-08
0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2.58E-10	9.82E-09	6.50E-09	7.45E-09	5.42E-09	1.04E-09	2.00E-10	3.85E-11	7.41E-12	1.18E-11
(Ci-sec/m^3)									(Ci-sec/m^3)
5.68E-17	3.81E-18	3.30E-18	5.35E-18	7.17E-18	4.03E-18	2.27E-18	1.28E-18	7.18E-19	3.11E-17
3.16E-07	2.72E-08	2.48E-08	4.35E-08	6.72E-08	4.69E-08	3.27E-08	2.28E-08	1.59E-08	3.13E-07
1.85E-01	1.89E-02	1.80E-02	3.33E-02	5.75E-02	4.69E-02	3.83E-02	3.13E-02	2.56E-02	3.01E-01
5.41E-24	2.94E-25	2.44E-25	3.73E-25	4.48E-25	2.12E-25	1.00E-25	4.73E-26	2.23E-26	1.94E-24
1.01E-10	7.81E-12	6.98E-12	1.18E-11	1.72E-11	1.10E-11	7.00E-12	4.46E-12	2.85E-12	7.71E-11
8.44E-02	8.61E-03	8.18E-03	1.52E-02	2.61E-02	2.13E-02	1.73E-02	1.41E-02	1.15E-02	1.36E-01
1.74E-01	1.76E-02	1.66E-02	3.07E-02	5.25E-02	4.24E-02	3.42E-02	2.75E-02	2.22E-02	2.72E-01
1.10E+01	1.12E+00	1.06E+00	1.97E+00	3.38E+00	2.75E+00	2.23E+00	1.81E+00	1.47E+00	1.76E+01
2.21E-03	2.08E-04	1.94E-04	3.50E-04	5.72E-04	4.33E-04	3.28E-04	2.49E-04	1.88E-04	2.81E-03

Dose Results

	Thyroid DCF (Rem/Ci)	Thyroid Dose (REM) 0 - 2 hr	Thyroid Dose (REM) 2 - 8 hr	Exposure 0-720 hrs (Ci-Sec/M3)	Skin DCF (REM-M3/ Sec-Ci)	Skin Dose (REM)	Body DCF (REM-M3/ Sec-Ci)	Body Dose (REM)
I 131	1.10E+06	5.25E+00	2.93E-01	1.45E-02	3.07E-02	4.45E-04	5.59E-02	4.08E-05
I 132	6.30E+03	3.89E-15	1.21E-16	1.84E-15	1.10E-01	2.02E-16	3.55E-01	3.28E-17
I 133	1.80E+05	5.63E-02	2.96E-03	9.49E-04	8.90E-02	8.44E-05	9.11E-02	4.34E-06
I 134	1.10E+03	0.00E+00	0.00E+00	0.00E+00	1.42E-01	0.00E+00	4.11E-01	0.00E+00
I 135	3.10E+04	8.00E-06	3.66E-07	7.78E-07	7.86E-02	6.11E-08	2.49E-01	9.73E-09
	TOTAL	5.30E+00	2.96E-01		TOTAL	5.30E-04	TOTAL	4.51E-05

	Skin DCF (REM-M3/ Sec-Ci)	NG Skin Dose (REM)		Body DCF (REM-M3/ Sec-Ci)	NG Body Dose (REM)	
KR 83M	0.00E+00	0.00E+00		1.27E-05	5.61E-23	
KR 85M	4.97E-02	3.13E-08		2.31E-02	7.31E-10	
KR 85	4.84E-02	2.35E-02		3.31E-04	8.08E-06	
KR 87	3.36E-01	2.47E-24		1.33E-01	4.92E-26	
KR 88	7.76E-02	1.38E-11		3.38E-01	3.02E-12	
XE131M	1.33E-02	2.93E-03		1.25E-03	1.39E-05	
XE133M	2.96E-02	1.32E-02		4.29E-03	9.61E-05	
XE133	9.67E-03	2.77E-01		4.96E-03	7.14E-03	
XE135M	2.14E-02	1.07E-04		6.37E-02	1.61E-05	
	TOTAL	3.17E-01	3.17E-01 (w/ iodine)	TOTAL	7.27E-03	7.32E-03 (w/ iodine)

E. Realistic Dose Estimate for the Individual Designated to Close the Equipment Hatch

As noted in reference 23 and supporting documentation, the thyroid dose to the individual(s) designated to close the equipment hatch may be excessively high if he were to remain in the containment with design basis FHA parameters. Thus a realistic estimate of the thyroid (and whole body and skin) dose to this individual is made based on the following parameters.

Power – reference 28 indicates that fuel drops from up to 20 feet are not expected to cause damage to a fuel assembly that result in radioactive releases. The FSAR (ref. 11a) indicates only the outer row of fuel pins might be assumed to fail, so the power is taken to be 2775 MWt / 157 assy / 17 rows = 1.04 MWt.

Decay Time – historically FNP has started fuel movement at about 150 hours due to limitations in spent fuel pool heat removal capability. To bound future improvements, the accident is assumed at 120 hrs.

Release Fraction - the average peaking factor for the core is 1.0. Per ref 23 the pool DF for elemental iodine is expected to be 500. Ref 11e indicates the expected gap fraction for I^{131} (by far the limiting isotope) is 2.2% as compared to 8% used in input file FNPGAP31. With a gap activity 99.75% elemental and .25% organic the expected iodine release fraction to the containment air will be (.9975/500 + .0025) x 2.2/8 = 1.236 x 10^{-3}. Although reference 11e indicates the expected gap fraction for noble gases (except Kr^{85}) is less than the noble gas release modeled, it is unchanged for this evaluation.

Containment Volume – Operation of the containment cooling system will quickly dilute the released activity throughout the free volume of the containment, so use 2 x 10^6 ft^3. For the whole body dose, the individual is assumed exposed to the free volume above the operating deck (2 x 10^6 ft^3) x (0.822), which results in a Geometry Factor of $1173 / [(2 \times 10^6)(0.822)]^{(0.338)} = 9.3$.

Purge Rate – a slower purge will maintain the released activity inside the containment longer. Use of the slow speed exhaust rate will maximize the operator dose. Note however, the containment purge radiation monitor will still rapidly reach the isolation setpoint, thus purge will be rapidly terminated and the operator dose will be calculated based on the initial activity released into the containment.

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ACTIVITY RELEASED TO ENVIRONMENT AND IN EACH NODE AT END OF... 2.778E-06  (HRS)
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ISO	NAM	ENV.	Contnmnt	CtrlRoom
I	131	1.887E-06	8.235E-01	9.950E-10
I	131	2.363E-06	1.031E+00	1.246E-09
I	132	8.056E-22	3.515E-16	4.248E-25
I	132	1.009E-21	4.402E-16	5.319E-25
I	133	6.967E-08	3.040E-02	3.673E-11
I	133	8.724E-08	3.807E-02	4.600E-11
I	135	1.480E-11	6.456E-06	7.801E-15
I	135	1.853E-11	8.085E-06	9.769E-15

I^{131} is clearly the limiting isotope. The total I^{131} inventory released into the containment is 1.9 Ci.

KR	83M	1.629E-23	7.107E-18	8.588E-27
KR	85M	5.409E-12	2.360E-06	2.852E-15
KR	85	6.327E-05	2.761E+01	3.336E-08
KR	87	5.578E-32	2.434E-26	0.000E+00
KR	88	3.131E-16	1.366E-10	1.651E-19
XE	131M	2.761E-05	1.205E+01	1.456E-08
XE	133M	4.697E-05	2.049E+01	2.476E-08
XE	133	3.402E-03	1.485E+03	1.794E-06
XE	135	1.810E-07	7.900E-02	9.546E-11

27.6 E6 µCi/2 E6 ft^3/28317 ml/ft^3 yields 4.9 E-4 µCi/ml which is greater than the containment purge radiation monitor setpoint; thus purge would be expected to automatically isolate. NOTE, use of Ref. 11e would increase the activity resulting in a faster response time.

Then the operator thyroid dose would be

$$1.9 \text{ Ci} / (2 \times 10^6/35.3) \text{ m}^3 \times 3.47 \times 10^{-4} \text{ m}^3/\text{second} \times 1.1 \times 10^6 \text{ REM/Ci inhaled}$$
$$= 1.28 \times 10^{-2} \text{ REM per second of exposure.}$$

Then the dose for 30 minutes would be 1.28×10^{-2} REM/second x 1800 seconds = 23.0 REM, and for a 50 REM limit the maximum stay time would be 3900 seconds or 65 minutes.

Skin and whole body doses (calculated similarly for 1 hour, shown below) clearly are not limiting.

ISOTOPE	Activity (Ci)	Conc. (Ci / m3)	Skin DCF (R-m3/s-Ci)	Skin Dose (R)	WB DCF (R-m3/s-Ci)	WB Dose (R)
I-131	1.90E+00	3.35E-05	3.07E-02	3.70E-03	5.59E-02	7.25E-04
Kr-85	2.76E+01	4.87E-04	4.84E-02	8.48E-02	3.31E-03	6.24E-04
Xe-131m	1.21E+01	2.13E-04	1.33E-02	1.02E-02	1.25E-03	1.03E-04
Xe-133m	2.05E+01	3.62E-04	2.96E-02	3.85E-02	4.29E-03	6.00E-04
Xe-133	1.49E+03	2.62E-02	9.67E-03	9.12E-01	4.96E-03	5.03E-02
			TOTAL	1.05E+00		5.23E-02

Assuming all activity is released during the transit time, the transit doses are also not limiting. NOTE that the operator cannot be exposed to 100 % of both the inside containment and outside containment activities since each is 100% of the release.

ISOTOPE	Release (Ci)	Skin DCF (R-m3/s-Ci)	Skin Dose (R)	WB DCF (R-m3/s-Ci)	WB Dose (R)	Thy. Dose (R)
I-131	1.90E+00	3.07E-02	4.911E-05	5.59E-02	8.943E-05	0.610644
Kr-85	2.76E+01	4.84E-02	0.0011248	3.31E-03	7.692E-05	
Xe-131m	1.21E+01	1.33E-02	0.0001355	1.25E-03	1.274E-05	
Xe-133m	2.05E+01	2.96E-02	0.0005109	4.29E-03	7.405E-05	
Xe-133	1.49E+03	9.67E-03	0.0120911	4.96E-03	0.0062018	
		TOTAL	0.0139114		0.006455	0.610644

ENCLOSURE 3

Joseph M. Farley Nuclear Plant Units 1 and 2
Response to Second Request for Additional Information Related to
Request to Revise Technical Specifications - Containment Equipment Hatch

Compact Disc (CD) Containing:

ARCON96 Input Data Files

ARCON96 Output Data Files

χ/Q Calculation