VIRGINIA ELECTRIC AND POWER COMPANY

RICHMOND, VIRGINIA 23261

October 17, 1995

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

Serial No.	95-506
NL&OS/JBL	R2
Docket Nos.	50-338
	50-339
License Nos.	NPF-4
	NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS CHANGES
TO ALLOW THE CONTAINMENT PERSONNEL AIRLOCK
DOORS TO REMAIN OPEN DURING REFUELING OPERATIONS

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company requests amendments, in the form of changes to the Technical Specifications and changes to Facility Operating License Numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively. The proposed changes will modify the Technical Specifications to allow both of the containment personnel airlock doors to remain open during refueling operations. The proposed changes will also delete the license condition referencing the analyses for limiting doses to the control room operators.

A discussion of the proposed changes is provided in Attachment 1. The discussion includes identification of several inconsistencies that were found between the as-built plant configuration and the Updated Final Safety Analysis Report and Technical Specifications Bases noted during a review of the licensing bases for the fuel handling accident. The discussion of changes provides resolution of these inconsistencies. The proposed operating license and Technical Specifications changes are provided in Attachment 2.

The proposed changes have been reviewed and approved by the Station Nuclear Safety and Operating Committee and the Management Safety Review Committee. It has been determined that the proposed changes do not involve an unreviewed safety question as defined in 10 CFR 50.59 or a significant hazards consideration as defined in 10 CFR 50.92. The basis for our determination that the changes do not involve a significant hazards consideration is provided in Attachment 3.

The proposed Technical Specifications changes would significantly facilitate conduct of refueling activities in the containment during the upcoming North Anna Unit 1 refueling outage. The planned start date for this outage is February 10, 1996.

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Therefore, Virginia Electric and Power Company requests approval of the proposed Technical Specifications changes by February 1, 1996.

This request is similar to amendments for several other nuclear power plants which the NRC has approved in 1994 and 1995. Should you have any questions or require additional information, please contact us.

Very truly yours,

James P. O'Hanlon

James P. O'Hanlon
Senior Vice President - Nuclear

Attachments

cc: U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, N.W.
Suite 2900
Atlanta, Georgia 30323

Mr. R. D. McWhorter
NRC Senior Resident Inspector
North Anna Power Station

Commissioner
Department of Radiological Health
Room 104A
1500 East Main Street
Richmond, Virginia 23219

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. P. O'Hanlon, who is Senior Vice President - Nuclear, of Virginia Electric and Power Company. He is duly authorized to execute and file the foregoing document in behalf of that Company, and the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 17th day of October, 1995.

My Commission Expires: May 31, 1998.

Vicki L. Hull
Notary Public

(SEAL)

ATTACHMENT 1

DISCUSSION OF CHANGES

VIRGINIA ELECTRIC AND POWER COMPANY

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1.0 INTRODUCTION

North Anna Power Station Technical Specifications require that one of the containment personnel airlock doors be closed during core alterations or movement of irradiated fuel in containment which results in cycling of the personnel airlock doors for each containment entry. Changes are being proposed to the Technical Specifications that will allow both doors to remain open during fuel movements as long as a designated individual is available to close one airlock door after the containment is evacuated in the event of a Fuel Handling Accident inside containment.

A new analysis of the dose consequences from a Fuel Handling Accident has been performed to support this proposed change. This new analysis makes the conservative assumption that all of the radionuclides released from the reactor cavity or fuel pool are released without credit for retention in the containment or iodine removal by the charcoal filter systems. The new analysis is applicable to either a Fuel Handling Accident inside containment or in the fuel building. This conservative evaluation uses new atmospheric dispersion factors (χ/Q values) for the Control Room and Exclusion Area Boundary calculated based on equations provided in PNL-10286, "Atmospheric Dispersion Estimates in the Vicinity of Buildings" (Reference 1). The results of this analysis indicate that the doses remain within regulatory limits and Standard Review Plan guidelines and that no significant hazards consideration is created by the proposed Technical Specifications changes.

While the analysis does not take credit for radionuclide retention in containment or iodine removal by the charcoal filters, the containment and fuel building ventilation system requirements will remain as currently specified in the Technical Specifications except that the containment personnel airlock will be manually closed in the event of a Fuel Handling

Accident inside containment. The proposed changes also include a modification to the bases of the Technical Specifications to clarify the emergency power system requirements relative to mitigation of the consequences of a Fuel Handling Accident. In addition, a clarification of our response to NRC Question 6.72 of the original FSAR is provided. This clarification corrects inconsistencies between the as-built configuration of the Fuel Building Ventilation System and our response regarding conformance with the recommendations in Regulatory Guide 1.52.

2.0 BACKGROUND

North Anna Technical Specifications section 3.9.4 requires that one of the containment personnel airlock doors be closed during core alterations or movement of irradiated fuel in containment. This requires cycling the personnel airlock doors for each containment entry. Frequent containment entries are required while core alterations or fuel movement is in progress and the resulting heavy use of the personnel airlock produces wear and high maintenance requirements. There could be a large number of personnel in containment during refueling operations and it may take several cycles of the airlock to evacuate personnel from containment if a Fuel Handling Accident were to occur. The time required for these cycling operations would increase personnel doses.

A change is being proposed to Technical Specifications section 3.9.4 to allow both doors to remain open during fuel movements or core alterations provided that one door is operable and an individual is available to close the airlock door after personnel are evacuated if a Fuel Handling Accident should occur. This would reduce the maintenance requirements for the airlock doors and the dose to personnel in containment in the event of a Fuel Handling Accident. The Technical Specifications change being proposed is similar to changes recently approved by the NRC for several other utilities with similar personnel airlock doors. The proposed changes also provide clarification of the design and licensing bases for systems required to mitigate the consequences of a Fuel Handling Accident. These clarifications are necessary as a result of inconsistencies identified during an internal Company review.

In addition, proposed changes to the facility operating license are requested to delete License Condition 2.G for Unit 1 and License Condition 2.I for Unit 2. These license conditions were issued by the NRC on February 18, 1990, in response to a proposed license amendment submitted by Virginia Power on March 1, 1989, as supplemented on December 22, 1989. This license amendment was proposed by Virginia Power to resolve an unreviewed safety question associated with control room ingress and egress after an accident. (Pursuant to 10 CFR 50.59(c), a license holder that desires to make a change that involves an unreviewed safety question shall submit an application for amendment of his license.) In this case, the original control room dose analysis had not accounted for the impact on control room doses of air infiltration resulting from control room personnel ingress and egress after an accident. Control room doses were reevaluated for the following five accidents including the impact of air infiltration to the control room from multiple ingress and egress; Loss of Coolant Accident, Main Steam Line Break, Fuel Handling Accident, Steam Generator Tube Rupture, and Locked Rotor Accident. The unreviewed safety question associated with this analysis was resolved by amending the North Anna facility operating licenses to reference the revised control room dose analysis submittal because no other changes to the license were required.

The proposed changes to the Technical Specifications to allow both airlock doors to remain open during refueling operations is supported by a new Fuel Handling Accident analysis including a revised assessment of control room doses. Based on the results shown for this revised Fuel Handling Accident analysis, the limiting doses to control room operators still result from the Main Steam Line Break and Steam Generator Tube Rupture as described in the facility license conditions. However, because these limiting doses are now described in the UFSAR and the NRC Safety Evaluation report, facility operating license conditions 2.G and 2.I are no longer necessary. The continued existence of these separate license conditions is needlessly restrictive in that subsequent changes to the accident analysis are not normally addressed by further amending the license. Revisions to the aforementioned design bases accident analyses deemed not to involve an unreviewed safety question are normally revised by the licensee pursuant to 10 CFR 50.59 and subsequently reflected by updating the FSAR. In addition, these license conditions are redundant to information contained, appropriately, in the North Anna UFSAR. Therefore, to eliminate duplication and remove unnecessarily restrictive license conditions, License Condition 2.G for Unit 1 and 2.I for Unit 2 are now proposed to be deleted.

3.0 TECHNICAL SPECIFICATIONS CHANGES

The proposed Technical Specifications changes are described below and apply to North Anna Units 1 and 2:

Technical Specification 3.9.4

This section will be changed to allow both containment personnel airlock doors to be open during core alterations or movement of irradiated fuel within containment provided that one personnel airlock door is operable, and that there is at least 23 feet of water over the top of the reactor pressure vessel flange during movement of irradiated fuel or at least 23 feet of water above the top of irradiated fuel assemblies within the reactor pressure vessel during core alterations excluding movement of fuel assemblies.

Technical Specification 4.9.4

The surveillance requirements for the containment building penetrations are revised to require that if both doors of the containment personnel airlock are open, then one door shall be verified operable (capable of being closed) prior to the start of and at least once per 7 days during core alterations or movement of irradiated fuel in the containment building.

Technical Specification 3/4.9.4

The refueling operations bases is being changed to define that operability of the containment airlock door requires that the door is capable of being closed, that the door is unblocked and no cables or hoses are being run through the airlock, and that a designated individual is continuously available to close the airlock door. This individual must be stationed near the airlock.

Technical Specification 3/4.9.12

The refueling operations bases is also being changed to state that the operability of the fuel building ventilation system provides additional conservatism compared with the assumptions of the accident analyses.

Technical Specifications 3/4.8.1 And 3/4.8.2

The bases for the A.C. and D.C. Power Source Distribution Specification is being changed to clarify that one train of A.C. and D.C. busses must be available during fuel movement to ensure that the Control Room emergency ventilation system is operable in the event of a Fuel Handling Accident.

Facility Operating Licenses

The facility operating licenses (NPF-4 for Unit 1 and NPF-7 for Unit 2) will be changed to delete paragraph 2.G for Unit 1 and paragraph 2.I for Unit 2.

4.0 SAFETY SIGNIFICANCE

As shown in the analysis below, the containment and fuel building ventilation systems are not required to function to meet the regulatory limits or Standard Review Plan Guidelines for doses after a Fuel Handling Accident. However, in accordance with the requirements of the Technical Specifications and station operating procedures, these containment and fuel building ventilation systems are expected to be available to further reduce the consequences of a Fuel Handling Accident. The Control Room habitability systems are required to meet General Design Criteria 19 and will be available as described in current Technical Specifications to mitigate the radiation doses to Control Room operators. The only change to the design and operating requirements proposed is to allow both containment personnel airlock doors to remain open during core alterations or fuel movement in containment.

Should a Fuel Handling Accident occur within the containment, radiation monitors will automatically isolate the containment purge supply and exhaust. In addition, the Control Room operators will manually isolate the Control Room and initiate the bottled air supply. However, since both containment personnel airlock doors remain open until the containment can be evacuated, some release may occur through the airlock to the Auxiliary Building. Note that this release is expected to be small because there would be no significant differential pressure to force air from the containment. The small release to the Auxiliary Building may escape to the environment unfiltered since no ventilation system requirements are specified. The safety evaluation supporting this proposed change will assume the entire radioactive material release from the refueling cavity water to the containment air space is discharged through the ventilation stacks with no credit for isolation or iodine filtration. While this assumption is conservative with respect to the design and operating requirements, it demonstrates that neither non-safety related equipment nor manual action to close an airlock door is required to mitigate the consequences of a Fuel Handling Accident in order to meet regulatory limits or guidelines.

The safety evaluation includes a new analysis of the Fuel Handling Accident applicable to both inside containment and in the Fuel Building. Input assumptions for this dose evaluation are consistent with the guidelines given by the NRC Standard Review Plan (NUREG-0800, Reference 5) and Regulatory Guide 1.25 (Reference 6). Revised control room and Exclusion Area Boundary (EAB) atmospheric dispersion estimates (χ/Q) were calculated for these dose analyses based on North Anna meteorology data and equations for calculating χ/Q provided in PNL-10286, "Atmospheric Dispersion Estimates in the Vicinity of Buildings" (Reference 1). The dose calculations were performed with the LOCADOSE computer code (References 2 to 4). Dose consequences have been evaluated based on conservative assumptions for a Fuel Handling Accident inside containment or in the fuel building. The analysis methodology, assumptions, and results are described below.

4.1 Analysis Of χ/Q For The Control Room And EAB

4.1.1 Data For χ/Q Analysis

Atmospheric dispersion factors (χ/Q) relate the concentration of radionuclides χ (curies per cubic meter) at a receptor point to the radionuclide release rate Q (curies per second). The χ/Q values depend on the distance from release to receptor, building area and meteorology data. The meteorology data used were the hourly averages collected at the North Anna meteorology tower during the time period from January 1989 to December 1993, inclusive. The release to receptor distance used to determine control room χ/Q values was the minimum "stretched string" distance (that is the distance that airborne radioactive material would travel from the release point to the control room intake) from the ventilation stacks to the control room emergency intakes. This distance is 31.9 meters. The EAB distance modeled was 500 meters which is conservative compared with the actual 1500 meter distance to the EAB and is within the 8 to 1200 meter distance range for the experimental data used to evaluate the models in PNL 10286.

The air ventilation stacks are located near the turbine building which has an average 3229 m^2 surface area. This turbine building surface area was used in the equations shown below to determine Control Room χ/Q values. The EAB χ/Q was determined with the smaller containment cross section area of 1519 m^2 which results in a slightly more conservative EAB χ/Q that can be used both for a Fuel Handling Accident and for other accidents with releases near the containment.

4.1.2 <u>Assumptions And Methods Of Analysis For χ/Q</u>

The analysis to determine EAB χ/Q was based on the methodology provided in NRC Regulatory Guide 1.145 (Reference 7) as modified by equations for calculating χ/Q provided in PNL-10286, "Atmospheric Dispersion Estimates in the Vicinity of Buildings" (Reference 1). The analysis to determine Control Room χ/Q values was also based on the equations provided in PNL-10286. The Control Room χ/Q used for the 0-8 hour period is the χ/Q that bounds 95% of the values calculated for the five years of hourly data considered. Based on Murphy and Campe (Reference 8), χ/Qs for time periods from 8 hours out to 30 days are determined that bound the following percentiles of hourly data: 8 to 24 hours, 90%; 24 to 96 hours, 80%; 96 to 720 hours, 60%. The following Murphy and Campe recommended factors were then used to reduce Control Room χ/Q values to account for the probability that the wind will not always be blowing from the release to the receptor point: 0 to 8 hours, 1; 8 to 24 hours, 0.88; 24 to 96 hours, 0.75; 96 to 720 hours, 0.50.

Reference 1 provides the following equation for evaluation of χ/Q for each hour of meteorology data:

$$\frac{\chi}{Q}=\frac{1}{\pi\Sigma_y\Sigma_z U}$$

where (Equation 5, Reference 1):

$$\Sigma_y=(\sigma_y^2+\Delta\sigma_{y1}^2+\Delta\sigma_{y2}^2)^{\frac{1}{2}} \qquad \Sigma_z=(\sigma_z^2+\Delta\sigma_{z1}^2+\Delta\sigma_{z2}^2)^{\frac{1}{2}}$$

and (Equations 14, 15, 16 and 17, Reference 1):

$$\Delta\sigma_{y1}^2=9.13\times10^5[1-(1+\frac{x}{1000U})\exp(-\frac{x}{1000U})]$$

$$\Delta\sigma_{z1}^2=6.67\times10^2[1-(1+\frac{x}{100U})\exp(-\frac{x}{100U})]$$

$$\Delta\sigma_{y2}^2=5.24\times10^{-2}U^2A[1-(1+\frac{x}{10\sqrt{A}})\exp(-\frac{x}{10\sqrt{A}})]$$

$$\Delta\sigma_{z2}^2=1.17\times10^{-2}U^2A[1-(1+\frac{x}{10\sqrt{A}})\exp(-\frac{x}{10\sqrt{A}})]$$

In the equations above, x is the source to receptor distance, U is the wind speed, A is the building area and σ_y and σ_z are as defined by the curves given in Reference 7.

These equations were used to evaluate χ/Q for each hour with the North Anna meteorology data. These calculated χ/Q values were then sorted and the value found that bounds χ/Q for 95% of the hours evaluated. This 95% bounding χ/Q is then used for the 0 to 8 hour period after an accident for the Control Room. The 90%, 80% and 60% bounding χ/Q values are also determined to find the χ/Q values appropriate for time periods up to 30 days after an accident.

For the EAB, the one hour χ/Q values are also sorted into 16 sectors by wind direction. Each of these sectors represents a 22.5 degree compass point. The EAB distance used was 500 meters for all of the 16 sectors. This distance is conservative compared with the 1500 meter distance from containment to the EAB and is well within the 8 to 1200 meter distance range of experimental data used to evaluate the models in PNL 10286. The χ/Q values for each of the 16 sectors at the EAB are also sorted and the χ/Q is found that bounds all but 0.5% of the total number of hours of meteorology data evaluated. As specified in Reference 7, the maximum sector χ/Q is then compared with the 95% direction independent χ/Q and the larger of these two values is chosen as the 0-2 hour EAB χ/Q. Because EAB doses are evaluated for only the first 2 hours after an accident, EAB χ/Q values for longer time periods are not determined.

4.1.3 Results Of χ/Q Analysis

The limiting χ/Q for the North Anna EAB was determined with the methods discussed above to be 4.4×10^{-5} sec/m^3 for the South East sector. This maximum sector χ/Q was slightly higher than the 95% direction independent χ/Q. The Low Population Zone (LPZ) χ/Qs shown in the North Anna UFSAR were used for the LPZ:

LPZ χ/Q Values

Time Period	χ/Q
0 to 8 hr	1.1×10^{-5}
8 to 24 hr	7.3×10^{-6}
24 to 96 hr	3.0×10^{-6}
96 to 720 hr	8.2×10^{-7}

The Control Room χ/Q values calculated for the time periods up to 30 days were:

Control Room χ/Q Values

Time Period	χ/Q
0 to 8 hr	6.0×10^{-3}
8 to 24 hr	4.6×10^{-3}
24 to 96 hr	3.9×10^{-3}
96 to 720 hr	2.3×10^{-3}

4.2 Summary Of Fuel Handling Accident (FHA) Input Data And Assumptions

4.2.1 Fuel Handling Accident Description

A Fuel Handling Accident during refueling operations could release a fraction of the fission product inventory to the environment. An illustrative accident sequence consists of: the dropping of a fuel assembly, breaching of the fuel rod cladding, release of a portion of the volatile fission gases from the damaged fuel rods, absorption of some water soluble gases in and transport of the remainder of the soluble and all insoluble gases through the water to the air space over the water, possible air filtration prior to release into the environment, and dispersion of the released fission products into the atmosphere. The analysis presented below assumes that no filtration of iodine occurs prior to release to the environment.

This assumption of an unfiltered release conservatively bounds the proposed Technical Specifications change to allow both containment personnel airlock doors to be open during core alterations or fuel movement inside containment. Any release which occurs through the airlock before one of the doors is closed may not be filtered. In addition, the assumption of a full unfiltered release demonstrates that operation of the fuel and containment building ventilation systems to filter or retain radioactive material is not required to mitigate the consequences of a Fuel Handling Accident to ensure that the doses remain below regulatory limits or Standard Review Plan guidelines. Thus, this conservative assumption further supports the bases for the current fuel and containment building ventilation systems since all of the equipment in these systems is not designed to meet safety related criteria. It should be noted however, that operation of the fuel building ventilation system in accordance with Technical Specifications will result in a filtered rather than an unfiltered release. In addition, for a Fuel Handling Accident inside containment, radiation monitors are provided to automatically isolate the containment purge and terminate the release once the airlock door is manually isolated after containment evacuation. Under these conditions, the doses would be substantially reduced from those shown in the conservative analysis presented below.

To determine the quantity of radioactive material available for release, it is conservatively assumed that the fuel assembly with the peak fission product inventory is the one damaged. This inventory is based on maximum full power operation at the end of core life immediately preceding shutdown with a conservative radial peaking factor applied to all fuel rods in the assembly. A conservatively large fraction of volatile fission products is assumed to have migrated from the fuel matrix to the gap and plenum regions of the fuel rods prior to the Fuel Handling Accident. This fraction of fission products is assumed to be immediately released to the water around the fuel.

4.2.2 <u>Fuel Handling Accident Analysis Assumptions</u>

The North Anna Fuel Handling Accident (FHA) analysis was performed consistent with the following requirements and assumptions provided in Regulatory Guide 1.25 (Reference 6):

(1) The accident occurs 100 hours after shutdown. North Anna Technical Specification 3.9.3 requires a minimum 150 hour period between the shutdown of a unit and initiation of fuel movement, so the use of a 100 hour time period is conservative. Radioactive decay of the fission product inventory during the 100 hour interval between shutdown and the assumed commencement of fuel handling is incorporated into the analysis.

(2) The minimum water depth above the damaged fuel rods is 23 feet as required by Technical Specifications.

(3) All of the gap activity in the damaged rods is released and consists of 10% of the total noble gases other than Kr-85, 30% of the Kr-85, and 10% of the total radioactive iodine in the rods at the time of the accident.

(4) The values assumed for individual fission product inventories are calculated assuming full power operation at the end of core life immediately preceding shutdown. A radial peaking factor of 1.65 was used.

(5) The iodine gap inventory is composed of 99.75% inorganic species and 0.25% organic species.

(6) The pool decontamination factors for the inorganic and organic iodine species are 133 and 1, respectively, giving an overall decontamination factor of 100 (i.e., 99% of the total iodine released from the damaged rods is retained by the pool water). This difference in decontamination factors for inorganic and organic iodine species results in the iodine above the fuel pool being composed of 75% inorganic and 25% organic species.

(7) The retention of the noble gases in the pool is negligible.

(8) The radioactive material that escapes from the pool to the fuel or containment building is released within a two hour time period.

In addition, it was conservatively assumed that there is no filtration or radionuclide retention by the Fuel Building or containment ventilation systems.

4.2.3 Determination of Activity Released

The core inventory was calculated for North Anna assuming operation at 102% of full power, in accordance with Regulatory Guide 1.49. The core inventory at shutdown for this analysis was determined using the curies per megawatt factors given in the LOCADOSE computer code system (References 2 through 4). The effects of a 100 hour period of decay on this core inventory were then assessed. This evaluation included the contribution of iodine and noble gases which result from decay of Te isotopes. The amount of radioactive material released in a Fuel Handling Accident is determined from this 100 hour core inventory assuming that all of the rods in one of the 157 fuel assemblies in the North Anna core are damaged. The resulting activities released to the fuel or containment buildings (depending on the location of the accident) are given in Table 1.

4.3 North Anna Fuel Handling Accident Inside Containment

4.3.1 LOCADOSE Model

The LOCADOSE computer code system was used to calculate doses for the Fuel Handling Accident inside containment. The model for this accident considered three distinct volumes: the environment, the containment building, and the control room. For conservatism and to maintain consistency in the analysis of a Fuel Handling Accident in the containment or the fuel building, the volume and exhaust flow rates used for the containment analysis was the more conservative fuel building volume of 1.6×10^5 ft^3 and exhaust flow rate of 35,000 CFM which results in a more rapid release. This flow rate is high enough to ensure essentially complete release over a two hour period consistent with Regulatory Guide 1.25. The Control Room volume for North Anna is 2.3×10^5 ft^3. The dose calculations considered the initial activity shown in Table 1.

4.3.2 Containment Ventilation System Operating Requirements

The containment building ventilation system will continue to be operated in compliance with Technical Specifications requirements to isolate the containment in the event of a Fuel Handling Accident. The only change will be that the personnel airlock would be manually isolated after containment evacuation rather than requiring that one door be closed at all times. Isolation by the non-safety containment ventilation system, including manual action to close one personnel airlock door, would provide additional reduction below the doses in the analysis shown below.

4.3.3 Control Room Ventilation System Operating Requirements

During core alterations or fuel movements inside containment, direct communication will be established between fuel handling personnel in containment and the Control Room. Upon verbal notification of a Fuel Handling Accident with the potential for radionuclide release or upon receipt of a high radiation signal from the containment radiation monitors, the Control Room will be manually isolated, and the bottled air supply initiated. It is estimated that up to a 2 minute delay can occur between detection of a high radiation level and isolation of the control room. However, the transit time for any released activity from the radiation detection point to the control room emergency ventilation system intake is expected to exceed 2 minutes. Therefore, control room isolation is modeled as occurring at the start of the accident. As shown in Table 2, the control room is supplied with bottled air for 1 hour after the start of a Fuel Handling Accident and then with filtered air at a flow rate of 1000 CFM with an iodine filtration efficiency of 95% for organic and inorganic iodine through the remainder of the 30-day dose calculation period. No credit is taken for operation of fan/filter units to provide recirculation of the control room air. The fan/filter unit which supplies the 1000 CFM of filtered intake is supplied by emergency power to ensure that GDC 19 limits are met.

4.3.4 Atmospheric Dispersion Factors, Occupancy Factors, and Breathing Rate

Any releases from the containment building (either from the purge or from the airlock) would exhaust to the atmosphere through one of the ventilation stacks. As described above, Control Room and EAB χ/Q values were determined for the Fuel Handling Accident releases from these stacks. The control room occupancy factors recommended by Murphy and Campe were incorporated into the dose calculations to reflect that personnel would not be exposed to the released activity continuously for the entire 30-day dose calculation period. The breathing rate used for the control room, EAB and LPZ dose calculations was 3.47×10^{-4} m^3/sec.

4.3.5 Results For A Fuel Handling Accident Inside Containment

The results of the dose calculation for a Fuel Handling Accident inside containment using the model and assumptions described above are summarized in Table 3. The calculated doses are less than the GDC-19 and 10 CFR 100 limits as shown in the table. The Standard Review Plan section 15.7.5 guideline for a Fuel Handling Accident is that the doses should be well within (<25%) the 10 CFR 100 limits. The EAB and LPZ doses also meet this guideline even assuming no retention of radionuclides within the containment. The proposed Technical Specifications changes require a designated individual be available to close the personnel airlock after containment evacuation. By closing the airlock and isolating containment purge, both iodine and whole body doses would be reduced below the values shown in the event of a Fuel Handling Accident inside containment.

Table 1

Activity Released By A Fuel Handling Accident Inside Containment

Isotope	Activity (Ci)
I-131 (Elemental)	4.193 E+02
(Organic)	1.398 E+02
I-132 (Elemental)	3.807 E+02
(Organic)	1.269 E+02
I-133 (Elemental)	4.848 E+01
(Organic)	1.616 E+01
I-135 (Elemental)	3.941 E-02
(Organic)	1.314 E-02
Kr-85	3.815 E+03
Xe-131m	7.851 E+02
Xe-133m	1.923 E+03
Xe-133	1.209 E+05
Xe-135m	9.038 E-01
Xe-135	2.564 E+02

Table 2
Control Room Ventilation Flow Rates

Description	Flow Rate (cfm)	Applicability
Unfiltered inleakage	10	0 to 30 days
Recirculation rate	0	0 to 30 days
Filtered Intake	1000	1 hour to 30 days

Table 3
Doses For A Fuel Handling Accident Inside Containment

Dose Type	Control Room (REM)	GDC-19 Limit (REM)
Thyroid	19	30 (equivalent to 5
Skin	<1	30 Rem Whole Body)
Whole Body	<1	5

Dose Type	EAB (REM)	10 CFR 100 Limit (REM)
Thyroid	12	300
Whole Body	<1	25

Dose Type	LPZ (REM)	10 CFR 100 Limit (REM)
Thyroid	3	300
Whole Body	<1	25

4.4 North Anna Fuel Handling Accident In The Fuel Building

4.4.1 LOCADOSE Model

The LOCADOSE computer code system was used to calculate the doses for the Fuel Handling Accident in the fuel building. The model for this accident considered three distinct volumes: the environment, the fuel building, and the control room. The spent fuel building volume is 1.6×10^5 ft^3 and the Control Room volume 2.3×10^5 ft^3. The dose calculations considered the initial activity shown in Table 4, and flow out of the fuel building into the environment at the normal 35,000 CFM fuel building ventilation rate. This flow rate is high enough to ensure essentially complete release over a two hour period consistent with Regulatory Guide 1.25.

4.4.2 Fuel Building Ventilation System Operating Requirements

North Anna Technical Specifications require that the air from the Fuel Building be exhausted through iodine filters during fuel handling. The current analysis for the Fuel Handling Accident in the spent fuel pool, which is documented in Section 15.4.5 of the UFSAR, takes credit for reduced doses resulting from iodine removal by these filters even though the fans and other ventilation equipment is not designated as safety related and seismic. Taking credit for this filtration was justified by an evaluation showing that Fuel Handling Accident doses would be below 10 CFR 100 limits without filtration.

During an internal Company review of the current licensing bases for the Fuel Handling Accident, it was determined that the Company's response to NRC Question 6.72 (section 6.2 of the UFSAR) regarding conformance with the recommendations in Regulatory Guide 1.52 could be interpreted to be inconsistent with the design of the

Fuel Building Ventilation System. Specifically, Regulatory Position C.2.h of that guide states that the power supplies and electrical distribution systems should be designed in accordance with IEEE-308, "Criteria For Class 1E Electrical Systems Of Nuclear Power Generating Stations." Our response to Question 6.72 stated that the Auxiliary Building Filtration System complies with this requirement. Contrary to this, the fans in the Fuel Building exhaust system (Regulatory Guide 1.52 defines the atmospheric cleanup system to include the fans), which draw air through the filter bank, are non-safety related and powered from non-safety related power. In addition, our response to Question 6.7 stated that the filtration system complied with Position C.2.c of Regulatory Guide 1.52. This position states that all components of the filtration system should be designated as Seismic Category 1 if failure of a component would lead to the release of significant quantities of fission products to the working or outdoor environment. However, the only portion of the Fuel Building Ventilation System of Seismic Category 1 design is that connected to the safety related filter assembly bypass and header system. The exhaust fans from the fuel building are specifically not included. Our response to Question 6.72 should have included notations clearly identifying these two exceptions to the recommendations of Regulatory Guide 1.52. As noted above, these exceptions are acceptable since prior analysis demonstrated that the resulting radiation doses with no filtration would still be significantly below the limits of 10 CFR 100.

The new analysis of the Fuel Handling Accident presented herein takes no credit for filtration. However, the Fuel Building exhaust will continue to be filtered by charcoal filters during fuel handling as required by the current Technical Specifications. The results of the new analysis as described below demonstrate that the resulting offsite doses are well within (<25%) of the Part 100 limits with no credit for filtration. Therefore, the new analysis continues to support the current design and operation of the Fuel Building Ventilation System. If filtration were considered in the analysis, charcoal filter iodine decontamination factors of 90% for inorganic iodine and 70% for organic iodine would reduce the calculated thyroid doses to 15% of the values reported below.

4.4.3 Control Room Ventilation System Operating Requirements

In accordance with the current design of the Control Room ventilation system, the North Anna control room will be automatically isolated, and the bottled air supply initiated, upon receipt of a high radiation signal from radiation monitors in the fuel building. Per design, a 2 minute delay will occur between detection of a high radiation level and isolation of the control room. However, the transit time for any released activity from the radiation detection point to the control room ventilation system intake will be more than 2 minutes. Therefore, control room isolation is modeled as occurring at the start of the accident. As shown in Table 5, the control room is supplied with bottled air for 1 hour after a Fuel Handling Accident and then with filtered air at a flow rate of 1000 CFM with an iodine filtration efficiency of 95% for organic and inorganic iodine through the remainder of the 30-day dose calculation period. No credit is taken for operation of fan/filter units to provide recirculation of the control room air. The fan/filter unit, which supplies the 1000 CFM of filtered intake, is supplied by emergency power to insure that the GDC 19 limits will be met.

4.4.4 Atmospheric Dispersion Factors, Occupancy Factors, and Breathing Rate

The fuel building ventilation system exhausts to the atmosphere through one of the ventilation stacks. As described above, control room and EAB χ/Q values were determined for the Fuel Handling Accident releases. The control room occupancy factors recommended by Murphy and Campe were incorporated into the dose calculations to reflect that personnel would not be exposed to the released activity continuously for the entire 30-day dose calculation period. The breathing rate used for the control room, EAB and LPZ dose calculations was 3.47×10^{-4} m^3/sec.

4.4.5 Pesults For A Fuel Handling Accident In The Fuel Building

The results of the dose calculation for a Fuel Handling Accident in the fuel building using the model and assumptions described above are summarized in Table 6. The calculated doses are less than the GDC-19 and 10 CFR 100 limits as shown in the table. The Standard Review Plan section 15.7.5 guideline for a Fuel Handling Accident is that the doses should be well within (<25%) the 10 CFR 100 limits. The EAB and LPZ doses also meet this guideline even assuming no iodine filtration by the fuel building ventilation system. Again any releases from the fuel building would be expected to pass through charcoal and particulate filters which would reduce the calculated thyroid doses to 15% of the values shown.

The new analysis for the Fuel Handling Accident shows that the components of the Fuel Building Ventilation System utilized to filter the exhaust are not required to mitigate the consequences of a Fuel Handling Accident to meet the regulatory limits or Standard Review Plan guidelines. This continues to support the position that these components do not have to be designated as safety related or supplied by emergency power. However, the Control Room emergency ventilation system is required to mitigate the consequences of a Fuel Handling Accident to meet GDC 19 limits and must be provided with emergency power. The bases section for Technical Specifications 3/4.8.1 and 3/4.8.2 currently indicates that emergency power is available to help recover from accidents such as a Fuel Handling Accident. This could be interpreted to imply that all systems potentially used to reduce the consequences of a Fuel Handling Accident are supplied by emergency power even if the systems are not required to mitigate accident consequences in order to meet regulatory limits or guidelines. Therefore, the bases section for Technical Specifications 3/4.8.1 and 3/4.8.2 is being clarified to indicate that emergency power is provided specifically to the control room ventilation system.

Table 4

Activity Released By A Fuel Handling Accident In The Fuel Building

Isotope	Activity (Ci)
I-131 (Elemental)	4.193 E+02
(Organic)	1.398 E+02
I-132 (Elemental)	3.807 E+02
(Organic)	1.269 E+02
I-133 (Elemental)	4.848 E+01
(Organic)	1.616 E+01
I-135 (Elemental)	3.941 E-02
(Organic)	1.314 E-02
Kr-85	3.815 E+03
Xe-131m	7.851 E+02
Xe-133m	1.923 E+03
Xe-133	1.209 E+05
Xe-135m	9.038 E-01
Xe-135	2.564 E+02

Table 5
Control Room Ventilation Flow Rates

Description	Flow Rate (cfm)	Applicability
Unfiltered inleakage	10	0 to 30 days
Recirculation rate	0	0 to 30 days
Filtered Intake	1000	1 hour to 30 days

Table 6
Doses For Fuel Handling Accident In The Fuel Building

Dose Type	Control Room (REM)	GDC-19 Limit (REM)
Thyroid	19	30 (equivalent to 5
Skin	<1	30 Rem Whole Body)
Whole Body	<1	5

Dose Type	EAB (REM)	10 CFR 100 Limit (REM)
Thyroid	12	300
Whole Body	<1	25

Dose Type	LPZ (REM)	10 CFR 100 Limit (REM)
Thyroid	3	300
Whole Body	<1	25

5.0 SUMMARY AND CONCLUSIONS

Technical Specifications changes are proposed which will allow the both containment personnel airlock doors to remain open during core alterations or fuel movements in containment provided that one door is operable and capable of being closed. To support the proposed change, the Fuel Handling Accident applicable to both the containment and Fuel Buildings have been reanalyzed using new methods and conservatively assuming a full release to the environment without credit for ventilation system filtration or retention of radionuclides in the containment or Fuel Buildings. The results of the new analysis demonstrated that radiation doses would be well within (<25%) the 10 CFR 100 limits and meet the General Design Criteria 19 limits. Therefore, it is concluded that operation of North Anna Units 1 and 2 consistent with the Technical Specifications changes proposed herein will be acceptable and meet all regulatory requirements.

6.0 REFERENCES

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