March 8, 2002

Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060-6711

SUBJECT: SURRY UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: ALTERNATIVE SOURCE TERM (TAC NOS. MA8649 ANDMA8650)

Dear Mr. Christian:

The Commission has issued the enclosed Amendment No. 230 to Facility Operating License No. DPR-32 and Amendment No. 230 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments change the Technical Specifications (TS) in response to your application transmitted by letter dated April 11, 2000, as supplemented by letters dated August 28 and November 20, 2000, and April 11, July 31, November 19, and December 20, 2001, and February 8, 2002.

These amendments revise the TS requirements to be consistent with an alternative source term in accordance with the requirements of 10 CFR 50.67, "Accident Source Term."

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

Sincerely,

/RA/

Gordon E. Edison, Sr. Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosures:
1. Amendment No. 230 to DPR-32
2. Amendment No. 230 to DPR-37
3. Safety Evaluation

cc w/encls: See next page

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VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 230
License No. DPR-32

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

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated April 11, 2000, as supplemented by letters dated August 28 and November 20, 2000, and April 11, July 31, November 19, and December 20, 2001, and February 8, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-32 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 230 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by Leonard N. Olshan for/

Richard J. Laufer, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 8, 2002

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 230
License No. DPR-37

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

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated April 11, 2000, as supplemented by letters dated August 28 and November 20, 2000, and April 11, July 31, November 19, and December 20, 2001, and February 8, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-37 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 230 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by Leonard N. Olshan for/

Richard J. Laufer, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 8, 2002

ATTACHMENT TO

LICENSE AMENDMENT NO. 230 TO FACILITY OPERATING LICENSE NO. DPR-32

LICENSE AMENDMENT NO. 230 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

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

Remove Page	Insert Page
TS 3.4-3	TS 3.4-3
TS 3.7-27	TS 3.7-27
TS 3.8-5	TS 3.8-5
TS 3.10-1	TS 3.10-1
TS 3.10-2	TS 3.10-2
TS 3.10-3	TS 3.10-3
TS 3.10-4	TS 3.10-4
-------	TS 3.10-4a
TS 3.10-5	TS 3.10-5
TS 3.10-6	TS 3.10-6
-------	TS 3.10-6a
TS 3.10-7	TS 3.10-7
TS 3.19-2	TS 3.19-2
TS 3.22-1	TS 3.22-1
TS 3.22-2	TS 3.22-2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 230 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 230 TO FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

By letter dated April 11, 2000, supplemented by submittals dated August 28 and November 20, 2000, and April 11, July 31, November 19, and December 20, 2001, and February 8, 2002, Virginia Electric and Power Company (the licensee) requested amendments to Facility Operating License Nos. NPR-32 and NPR-37 for the Surry Power Station Units 1 and 2 (Surry). The supplements contained clarifying information only and did not change the initial no significant hazards consideration determination or expand the scope of the initial application.

The licensee submitted these license amendments as a full-scope implementation of the alternative source term (AST) pursuant to 10 CFR 50.67, “Accident Source Term,” as described in Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.” For this license amendment request implementing the full-scope AST, the licensee re-analyzed and submitted the radiological consequences for two design basis accidents (DBAs): the loss-of-coolant accident (LOCA) and the fuel handling accident (FHA).

The licensee requested these license amendments based on the participation of Surry as a leading pilot plant in the development of 10 CFR 50.67, RG 1.183, and Standard Review Plan (SRP) 15.01, “Radiological Consequence Analyses Using Alternative Source Terms” as a member of the Nuclear Energy Institute (NEI) Task Force. In 1998, the staff used Surry as a rebaselining plant for evaluating the impact of implementing the AST at operating nuclear plants as described in SECY 96-242, “Use of the NUREG-1465 Source Term at Operating Reactors.” The results and findings from the rebaselining study for implementing the AST at operating reactors (the Grand Gulf, Surry, and Zion stations) were provided in SECY 98-154, “Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors.”

In these license amendments, the licensee requested that:

a. Technical Specification (TS) 3.4 Basis, “Spray Systems,” TS 3.8 Basis, “Containment,” and TS 3.19 Basis, “Main Control Room Bottled Air System,” be amended to allow a slight atmospheric pressure (0.5 psig) in containment during 1 to 4 hours following a LOCA, with subatmospheric pressure being reached within 4 hours.

b. TS Table 3.7-5, "Automatic Functions Operated from Radiation Monitors Alarm," be amended to delete the automatic function requirements and setpoint for the containment particulate and gas monitors, as well as the manipulator crane area monitors.

c. TS 3.10, "Refueling," and its Basis be amended to revise the Applicability and Objective statements to also include irradiated fuel movement in the fuel building.

d. TS 3.10, "Refueling," and its Basis be amended to delineate which conditions apply during refueling operations or during irradiated fuel movement in the fuel building.

e. TS 3.10, "Refueling," and its Basis be amended to revise the requirements for the equipment access hatch, the personnel airlock, and penetrations having a direct path to the outside atmosphere to be capable of being closed.

f. TS 3.10, "Refueling," and its Basis be amended to delete the requirement for testing and operability of the containment purge system and automatic isolation of this system during refueling.

g. TS 3.10, "Refueling," and its Basis be amended to revise the requirement for operability and continuous monitoring of the manipulator crane area monitors, containment particulate and gas monitors, fuel pit bridge radiation area monitor, and ventilation Vent Stack 2 particulate and gas monitors for identification of the occurrence of an FHA.

h. TS 3.10, "Refueling," and its Basis be amended to delete the requirement to filter fuel building exhaust during refueling.

i. TS 3.10, "Refueling," and its Basis be amended to delete the requirement to filter containment purge exhaust during refueling.

j. TS 3.10, "Refueling," be amended to add requirements to have two trains of the control room bottled air system operable during refueling operations and during irradiated fuel movement in the fuel building, as well as actions for its operability. (This requirement parallels the existing emergency ventilation system requirement in TS 3.10.)

k. TS 3.10, "Refueling," be amended to clarify the requirement to cease refueling operations if the limiting conditions are not met.

l. TS 3.22, "Auxiliary Ventilation Exhaust Filter Trains," and its Basis be amended to delete the requirement to manually realign auxiliary ventilation from the refueling mode on a safety injection signal (applicable for all fuel handling instead of decayed fuel only).

2.0 EVALUATION

The licensee re-analyzed and submitted the radiological consequence assessment for a postulated LOCA to support Amendment Item "a" above, and for a postulated FHA to support requested Amendment Items "b" through "l" above.

2.1 Loss-of-Coolant Accident

The current radiological consequence analysis for the postulated LOCA is based on the accident source term described in Technical Information Document (TID)-14844 and it is provided in Surry Updated Final Safety Analysis Report (UFSAR) Section 15.6. To demonstrate that the Surry Units 1 and 2 engineered safety features (ESFs) designed to mitigate the radiological consequences will still remain adequate after implementing the changes requested in this license amendment, the licensee reevaluated the offsite and control room radiological consequences of the postulated LOCA. The licensee has implemented an AST in this reevaluation. The licensee submitted the results of its offsite and control room dose calculations and provided the fission product transfer and removal models along with the major assumptions and parameters used in its dose calculations.

As documented in the submittals, the licensee determined that after the implementation of the changes requested in this license amendment with use of an AST, the existing ESF systems at Surry Units 1 and 2 will still provide assurance that the total radiological consequences of the postulated LOCA at the exclusion area boundary (EAB), in the low population zone (LPZ), and in the control room meet the acceptable radiation dose criteria. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criteria of 10 CFR 50.67 (a TEDE of 25 rem for EAB and LPZ and a TEDE of 5 rem for control room operator) replace the previous whole body and thyroid dose guidelines of 10 CFR 100.11 (300 rem to the thyroid and 5 rem to the whole body at EAB and LPZ) and General Design Criteria (GDC) 19 (5 rem to the whole body). To verify the licensee's determination, the staff reviewed the licensee's analyses and performed independent confirmatory radiological consequence dose calculations for the following three potential fission product release pathways after the postulated LOCA:

(1) containment leakage,

(2) leakage from ESF systems outside containment, and

(3) emergency core cooling system (ECCS) back-leakage to Refueling Water Storage Tank (RWST) through check valves in the normal suction line from the RWST.

2.1.1 Containment Leakage

Surry Units 1 and 2 have a subatmospheric primary containment design. This means the normal operating containment pressure is less than atmospheric pressure and, following a LOCA, the containment systems are designed to return the containment pressure to less than atmospheric pressure within a specified time interval. The current primary containment design basis acceptance criteria for Surry Units 1 and 2 following a LOCA for containment integrity and containment leak rate are: (1) the containment peak pressure must be less than 45 psig, (2) the containment must be depressurized within 1 hour to less than atmospheric pressure, and (3) the calculated peak containment pressure after 1 hour must be less than 0.0 psig. The licensee proposes to change TS Basis Sections 3.4, "Spray Systems," 3.8.D, "Containment Internal Pressure," and 3.19, "Main Control Room Bottled Air System," to revise the second and third criteria to state that, for radiological dose calculations, acceptable results are obtained with a positive pressure of 0.5 psig in containment during the 1- to 4-hour interval following a LOCA, with subatmospheric pressure being reached within the containment at 4 hours and maintained subatmospheric thereafter.

Appendix A of RG 1.183 states that the primary containment should be assumed to leak at the peak TS leak rate, L_a. Contrary to this guidance, the licensee has assumed that leakage from the containment varies with containment pressure. The licensee assumed that the containment leak rate is maintained at L_a during the first hour following initiation of a LOCA. Then, the licensee calculated the containment leak rate as a function of containment pressure by assuming that containment leakage can be modeled as an orifice in incompressible flow. For the dose calculations, the licensee assumed that the containment pressure is 0.5 psig from 1 hour to 4 hours. The leak rate corresponding to 0.5 psig is obtained using the equation for incompressible orifice flow and is approximately 20 percent of L_a.

The staff disagrees that primary containment leakage can be modeled as flow through an orifice. Leaks from containment do not behave in a regular, predictable way. Leak flow paths will not, in general, remain unchanged for the pressure and temperature ranges of a LOCA. Furthermore, the number of leaks may increase or decrease as the pressure varies. However, the staff considers a leak rate of 20 percent of L_a to be conservative for a containment pressure of 0.5 psig since the actual leakage path would be more tortuous than the flow through an orifice, and therefore has a higher flow resistance than flow through an orifice. The licensee's calculations also assumed that the temperature of the containment atmosphere remains constant as the pressure decreases. This is conservative since the actual temperature of the containment atmosphere would be reduced by the action of passive heat sinks and the spray and a lower temperature would result in a low flow rate. Therefore, the leakage value the licensee has used for the dose analysis, 20 percent of L_a is conservative and acceptable.

The licensee has not proposed any changes to the primary containment structure, heat removal systems, containment integrity (peak pressure and temperature) accident analyses, containment leak testing, or the TS associated with any of these. The changes in containment leak rate assumptions are only for offsite control room dose analyses. Note that the design basis LOCA analysis for determining peak containment pressure and temperature demonstrates that a containment pressure of 0.0 psig is reached within 1 hour. Therefore, the assumption made for dose analysis is conservative in comparison to the design-basis containment LOCA analysis with respect to containment leakage.

Although the leakage rate used in the LOCA dose calculations varies with time, the licensee will still be required to leak test the containment and demonstrate that the containment leakage is less than the maximum allowed containment leakage, $L_{a,}$ at the peak LOCA pressure, P_a, as specified in 10 CFR Part 50 Appendix J Option B.

During a LOCA, radioactive iodine is released to the containment from damaged fuel and some of it may leak out to the environment, producing offsite radiation doses. In order to minimize these doses, the iodine has to be removed from the post-accident containment atmosphere. There are two containment spray systems inside the Surry containment: the containment spray system with two separate pump trains, and the recirculation spray system with two inside and two outside recirculation spray pump trains. The licensee assumed that based on a single-failure criterion, one containment spray pump train and one train each in the inside and outside recirculation pump trains are operational. The licensee reanalyzed removal of iodine from the containment and reported the results. In the analysis, the licensee made a conservative assumption that iodine is removed by containment sprays only and iodine removal by deposition on containment walls is disregarded. The rates at which iodine is removed by sprays are determined by the iodine removal coefficients (λ) and by the maximum amount of iodine which can be removed from the containment atmosphere by sprays by the limiting decontamination factor (DF). The licensee calculated these parameters and reported them.

There are two types of iodine removal coefficients: λ_p for particulate iodine and λ_e for elemental iodine. Since the removal mechanisms for these two forms of iodine are different, two different methods are used for determination of the λs. Iodine in the particulate form consists of iodine compounds, mostly cesium iodide, forming small aerosol particles. The licensee used the methodology from NUREG/CR- 5966, “A Simplified Model of Aerosols Removal by Containment Sprays,” to calculate removal coefficient λ_p for this form of iodine. Since the interaction between spray drops and aerosols is a function of spray drop height and spray water flux, the value of λ_p varies for spray headers situated at different locations in the containment. The λ_p for the whole containment is determined by adding λ_p for individual spray headers. In its submittal, the licensee tabulated the values of λ_p for different time intervals. It also included a sample calculation, which permitted the staff to evaluate the methodology. The staff found the determination of λ_p for particulate iodine satisfactory.

At a temperature existing in the post-accident containment atmosphere, elemental iodine remains in a gaseous form and the removal mechanism is controlled by the iodine transfer rate to the spray droplets. In calculating removal coefficients for this form of iodine, the licensee used the model from Section 6.5.2 of the SRP. In this model λ_e is a function of spray drop height and of spray water flux. Similarly as for particulate iodine, the licensee determined λ_e for the whole containment by adding λ_es calculated for individual spray headers. Using this method, the licensee calculated λ_e=35 hr^{-1}. However, in its later calculation of elemental iodine removal from the containment atmosphere, the licensee used the value of λ_e=10 hr^{-1}, which introduced a very significant conservatism in the calculation.

The time at which removal of elemental iodine stops occurs when the containment DF reaches its limiting value. Section 6.5.2 of the SRP specifies that this limiting value should not exceed 200. Therefore, in its calculation the licensee used DF=200, even though its actual calculated value for the limiting DF was higher. The licensee also made an assumption that the containment sprays start operating only after the end of iodine release. This assumption overestimated the time when iodine ceased to be removed. A more realistic assumption is that there is a time period when the release and removal of elemental iodine occurs simultaneously. The staff’s calculation has indicated that this will result in a slightly shorter time when DF=200 is reached and elemental iodine ceases to be removed. However, in both these cases, the total amount of iodine removed from the containment atmosphere by sprays remains the same. The staff finds, therefore, that the licensee’s method for calculating limiting amounts of iodine removed from the containment atmosphere will provide acceptable results.

The staff finds that the method and removal rates used by the licensee for determining the aerosol and elemental iodine removal rates are acceptable since they are consistent with the guidelines provided in RG 1.183 and SRP Section 6.5.2, respectively. The licensee used the containment air mixing rate of 2 unsprayed volumes per hour between the sprayed and unsprayed regions of the containment atmosphere, which is consistent with the guidelines provided in RG 1.183.

2.1.2 Post-LOCA Leakage From Engineered Safety Features Outside Containment

The leakage from ESF components will occur in two buildings at Surry Units 1 and 2: the safeguards building and auxiliary building. The licensee assumed the leakage into the safeguards building to start at 415 seconds following the postulated LOCA when the spray recirculation pumps start taking suction from the containment sump and to continue leaking at the same rate for the entire duration of the accident (30 days). The recirculation pumps are located in the safeguards building. The leakage is estimated to be 1928 cc/hr, which is 2 times the design basis leakage limit assumed in the Surry UFSAR. The licensee further assumed the leakage into the auxiliary building to start at 2300 seconds following the postulated LOCA when the safety injection system switches suction from the RWST to the containment sump and to continue to leak at the same rate for the entire duration of the accident. This leakage is estimated to be 7672 cc/hr, which is also 2 times the design basis leakage limit assumed in the Surry UFSAR.

The licensee assumed that 10 percent of the total iodine activity in the leaked fluid becomes airborne and is released directly to the environment throughout the entire period of the accident. The staff finds that the licensee’s leak rate assumptions and fission product transport models are acceptable since they are consistent with the guidelines provided in Section 5 of Appendix A to RG 1.183. The licensee has not requested, and the staff has not provided, any credit for holdup in the auxiliary building prior to release or any removal of fission products by the safety-related auxiliary building filtration system.

2.1.3 Emergency Core Cooling System Back-Leakage to Refueling Water Storage Tank

Following a postulated LOCA, the suction water source for the ECCS is switched from the RWST to the containment sump. In this configuration, check valves in the normal suction line from the RWST provide isolation between the containment sump and the RWST. The licensee assumed in its radiological consequence analysis that a 200 cc/min back-flow leakage occurs through check valves into the RWST based on the valve leakage measurements. The licensee further assumed that 10 percent of the total iodine activity in the leaked fluid becomes airborne and is released into the RWST air space. The RWST is an airtight tank and has a vent pipe that discharges into the safeguards building sump. The licensee assumed that the auxiliary ventilation system draws outside air into the RWST at a rate of 1000 cfm based on the system flow rate tests performed at Surry and that 1000 cfm of air is displaced through the vent pipe into the safeguards building. This air is then discharged to the environment through ventilation vent No. 2, which is located on the roof of the auxiliary building. The licensee did not take any credit for fission product removal by the auxiliary ventilation system filters and assumed it is released directly to the environment for the entire duration of the accident. The staff considers the licensee’s assumptions to be conservative and, therefore, acceptable.

2.1.4 Radiological Consequence of Loss-of-Coolant Accident

The staff reviewed the licensee’s fission product transport and removal models and the major parameters and assumptions used in its radiological dose consequence analyses and finds that they are acceptable. Although the staff performed its independent radiological consequence dose calculations as a means of confirming the licensee’s results, the staff’s acceptance is based on the licensee’s analyses. The results of the licensee’s radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee in its dose calculations and used by the staff for its confirmatory dose calculations are listed in Table 2. The radiological consequences at the EAB, at the LPZ, and in the control room, calculated by the licensee and by the staff, are all within the dose acceptance criteria specified in 10 CFR 50.67. Therefore, the staff concludes the license amendment requested to allow positive containment pressure for up to 4 hours instead of current limit of 1 hour is acceptable.

2.2 Fuel Handling Accident

The current radiological consequence analysis for the postulated design basis FHA is based on the accident source term described in TID-14844 and it is provided in Surry UFSAR Section 15.4.9. In this license amendment request, the licensee re-evaluated the radiological consequence resulting from a postulated FHA in the fuel building and in the containment with open personnel air lock and equipment access hatch. The licensee implemented the AST in its re-evaluation. The licensee concluded in the submittals that the radiological consequences resulting from the FHA in the fuel building and in the containment with open personnel air lock and equipment access hatch meet the dose acceptance criteria specified in SRP 15.0.1 for the EAB and LPZ and that specified in 10 CFR 50.67 for the control room operator.

The licensee reached this conclusion as a result of:

(1) implementing the AST,

(2) using an alternative to the fraction of iodine-131 (I-131) inventory in fuel gap provided in RG 1.183,

(3) using a fission product decay period of 100 hours (time period from the reactor shutdown to the first fuel movement),

(4) using the decontamination factors of 500 and 1 for the iodine isotopes in elemental and organic forms respectively, in the spent fuel pool or in the reactor cavity with minimum water depth of 23 feet,

(5) using the ARCON96 model documented in NUREG/CR-6331, Rev. 1, “Atmospheric Relative Concentrations in Building Wakes,” for determining the atmospheric dispersion factors at the control room air intake,

(6) not taking any credit for removing fission products by the safety-related high efficiency particulate air (HEPA) filters or charcoal adsorbers in the auxiliary ventilation exhaust filter trains prior to release to the environment from either containment or the fuel building during refueling operation, and

(7) eliminating automatic isolation capability for the containment ventilation purge system during refueling operation.

The licensee assumed that all of the 204 fuel rods in one fuel assembly have failed instantaneously, releasing all fission products in the fuel gap to the water surrounding the fuel assemblies. For determining I-131 inventory in the fuel gap available for release, the licensee proposed an alternative to the fraction of I-131 inventory in the fuel gap provided in RG 1.183. In determining the alternative fraction, the licensee considered the data and the method developed by the industry.

This method attempts to bound the fission product release fractions with a limit curve as a linear function of fuel burnup. The industry provided these data and the method to the staff during development of RG 1.183. The staff reviewed the data and determined that the data, by themselves, did not provide an adequate basis to accept the fission product inventory in the fuel gap proposed by the industry. This determination was based on data from fuel used in current reactors, which likely was not exposed to heating rates associated with non-accident power transients, and the historical data could have limited applicability to fuel used in current and more aggressive fuel management practices.

The licensee evaluated the data and the limit curve provided by the industry to ascertain whether it provided an appropriate bound for fission product release behavior of light-water reactor fuel operating under specific fuel management practices at Surry. The licensee concluded that the limit curve proposed by the industry does not bound the expected fission product inventory in the fuel gap for Surry based on the core management and rod power conditions at Surry. Therefore, the licensee developed its own data for determining the total amounts of fission product inventory, including I-131, in the fuel gap available for release using the methodologies provided by the industry.

First, the licensee calculated time-averaged linear heat generation rates (LHGR) for typical fuel assemblies discharged after three cycles of irradiation at Surry. This calculation is based on the core design predictions for four recent Surry fuel cycles and specific Surry fuel pellet design. Then, the licensee estimated the fractions of fission product in fuel gap using the Surry LHGR values. The staff finds that the data and the methodology used by the licensee for estimating the fractions of fission product in the fuel gap at Surry are acceptable.

In Table 3.2-2 of Attachment 2 (July 31, 2001 submittal), the licensee provided the estimated gap fraction values and their corresponding maximum fuel burnups for three core regions (once-burned, twice-burned, and thrice-burned) for each of four recent fuel cycles evaluated by the licensee. Based on the estimated gap fraction values in Table 3.2-2, and with some margin for future core design flexibility, the licensee proposed the following gap fraction values:

Once-burned gap fraction:	3%
Twice-burned gap fraction:	5.35%
Thrice-burned gap fraction:	6%

The licensee stated that the time-average LHGR is the key fuel rod parameter affecting fission gas release used in the FHA analysis. The licensee further stated that the constraint is specified in terms of LHGR and associated fission gas gap fractions since various combinations of assembly rod power and burnup history can satisfy the above values. The staff agrees with these statements. In a letter to NRC dated December 20, 2001, the licensee committed to confirm, during the course of performing reload design calculations for the Surry cores, that all fuel assembly locations will satisfy the above figures. The staff reviewed the licensee's proposed values and finds that they are acceptable for the postulated FHA at Surry. The staff's acceptance of the alternative gap fractions proposed by the licensee is based on: (1) the licensee reducing the uncertainty of estimating gap release by performing site-specific analyses using the Surry fuel design and its fuel management practices, and (2) the licensee using methods for estimating gap releases acceptable to the staff.

The staff's acceptance of alternative gap fractions is limited to the postulated FHA at Surry only and the staff will review the gap fractions for other non-LOCA design basis accidents each on a case-by-case basis.

The current TS require that the equipment hatch and at least one air lock door in the personnel air lock be "properly" closed during refueling operations. This is consistent with the current analysis of the FHA. The licensee is proposing to revise this requirement to require that the equipment hatch door and the air lock door be capable of being closed. This is consistent with the licensee's proposed AST analysis of the FHA. The licensee's April 11, 2000, letter states that the licensee is not proposing a time interval for closure of the equipment hatch and the air lock door since the revised FHA analysis assumes leakage from these flow paths with acceptable consequences. This is an exception to the 30-minute closure time guidance of RG 1.183. In addition, the licensee stated that closure will be accomplished only as allowed by containment dose rates to prevent excessive radiological dose to plant workers since attaching the equipment access hatch door requires a containment entry.

In an April 11, 2001, response to a staff request for additional information, the licensee clarified the earlier description by stating that procedures will be followed for fuel handling that are similar to those used for response to a loss-of-decay heat event. These include keeping a log to identify and track containment openings, designating members of a team to close openings, and pre-staging needed equipment. The licensee further stated that "assuming acceptable radiological protection conditions exist, containment closure will be established within 45 minutes following the decision to isolate containment." The staff considers the difference between the RG 1.183 guidance of a 30-minute closure time and the 45 minutes proposed by the licensee to be acceptable because the licensee's calculations demonstrate that the offsite dose limits are not exceeded without closure of the containment (see Section 2.1 of this Safety Evaluation).

The licensee states that the penetrations that are allowed to be open are those that provide a direct path from the containment atmosphere to the outside atmosphere. For these penetrations, which provide a direct path from the containment atmosphere to the outside atmosphere, the proposed TS will require that containment isolation valves shall be operable or the penetration shall be closed by a valve, blind flange, or equivalent or the penetration must be capable of being closed.

GDC 64, "Monitoring Radioactive Releases," requires monitoring effluent discharge paths during normal operation, anticipated occurrences, and accidents. In the April 11, 2001, response to a request for additional information, the licensee stated that in the event of an FHA, although there is no permanently installed radiation monitoring equipment in the proximity of the equipment hatch, the permanently installed radiation monitors elsewhere in the containment will provide information to assess the radiological conditions and in the case of a confirmed outflow, a radiological assessment will be conducted in the area of the equipment hatch. The licensee therefore complies with the requirements of GDC 64.

The staff reviewed the licensee's methods, parameters, and assumptions used in its radiological dose consequence analyses. To verify the licensee's radiological consequence assessments, the staff performed confirmatory radiological consequence dose calculations for the most limiting fuel drop scenario that will produce the most radiological consequence.
The scenario involves the drop of an irradiated fuel assembly onto the core in the containment, releasing all the gap activity in the damaged rods into the reactor cavity pool water while the

equipment hatch, personnel airlock doors, and other containment penetrations are open. Furthermore, the staff assumed that all releases are through the personnel airlock which provides the most conservative radiological consequence with highest atmospheric dispersion factors to the control room compared to those through the containment equipment hatch or a release from the fuel building.

Although the staff performed its confirmatory dose calculations, the staff's acceptance is based on the licensee's analyses. The results of the licensee's radiological consequence calculations are provided in Table 1, and the major parameters and assumptions used by the licensee in its dose calculations and by the staff for its confirmatory dose calculations are listed in Table 3. The radiological consequences at the EAB, at the LPZ, and in the control room, calculated by the licensee and by the staff, are all within the dose criteria specified in 10 CFR 50.67 (5 rem TEDE in the control room), and meet the dose acceptance criteria specified in the SRP 15.0.1 (6.3 rem TEDE at EAB and LPZ). Therefore, the staff finds the licensee's requested Amendment Items "b "through" I" to be acceptable.

2.3 Control Room Habitability

The Surry control room habitability system includes a compressed breathing air system (CBAS) and an emergency filtered air system. Following a DBA, the normal outside air supply and exhaust are isolated and the CBAS is activated on a safety injection signal for the postulated LOCA or manually activated from the control room for the postulated FHA. The CBAS is provided to maintain a positive control room pressure to assure outward leakage preventing unfiltered air inleakage into the control room. The CBAS consists of two banks of air bottles. Each bank is designed to provide 1 hour of positive pressure with 18,000 cubic feet of free air.

The control room operators manually activate the control room emergency filtered air system (CREFAS) 60 minutes or more after the initiation of the accident but prior to the depletion of the CBAS. The CREFAS maintains a positive control room pressure upon depletion of the bottled air supply. Four 100%-capacity fans, each with a rated capacity of 1000 cfm, are provided in parallel to provide emergency ventilation. One fan is used to supply emergency ventilation to each of the main control rooms and each of the emergency switchgear rooms. This system takes suction from the turbine building through a HEPA filter and charcoal adsorber. The staff used 90-percent iodine removal efficiency for charcoal adsorbers and 99-percent HEPA filter efficiency in its radiological consequence analyses.

The licensee reevaluated the control room habitability with the application of the AST and concluded that the radiological consequences to the control room operator resulting from the postulated LOCA and FHA are within the 5 rem TEDE criterion specified in 10 CFR 50.67 using up to a 500 cfm unfiltered air inleakage rate. In a response to the staff's request for additional information concerning control room unfiltered air inleakage, the licensee responded in its letter dated July 31, 2001, that the maximum allowable control room unfiltered air inleakage would be 500 cfm, thus meeting the dose limit specified in 10 CFR 50.67 for the control room operator. The staff verified the licensee's response with its confirmatory dose calculation using the 500 cfm unfiltered air inleakage rate for meeting the dose criterion.

The staff is currently working toward resolution of generic issues related to control room habitability, with a particular focus on the validity of the control room unfiltered air inleakage rates that are commonly assumed in licensees' analyses of control room habitability. Recent testing by 25 percent of current operating plants has shown that, in most cases, the measured unfiltered air inleakage rates exceeded the value assumed in the design basis analyses. While in each case the affected licensee was able either to reduce the excessive unfiltered inleakage or show the acceptability of the observed unfiltered inleakage rate, the collective experience caused concerns regarding those facilities that have not performed the enhanced inleakage testing.

The staff has determined that there is reasonable assurance that the Surry control room will be habitable with up to 500 cfm unfiltered air inleakage during the postulated accidents and this amendment may be approved before the resolution of this generic issue. The staff bases this determination on: (1) the maximum allowable unfiltered air inleakage rate of 500 cfm; (2) conservative assumptions and parameters used in the radiological consequence analyses; and (3) the low probability of the postulated accident, occurring during this interim period, that could result in radioactivity releases sufficient to challenge the ability of control room operators to protect the health and safety of the public. The approval of this amendment does not exempt Surry from regulatory actions that may be imposed in the future as this generic issue is resolved.

To verify the licensee's radiological consequence assessments, the staff performed confirmatory radiological consequence dose calculations for the control room operator. Although the staff performed independent radiological consequence dose calculations for the control room operator as a means of confirming the licensee's results, the staff's acceptance is based on the licensee's analyses. The results of the licensee's radiological consequence calculation for a control room operator are provided in Table 1, and the major parameters and assumptions used by the licensee in its dose calculations and by the staff for its confirmatory dose calculations are listed in Tables 2 and 3. The radiological consequences for control room operators calculated by the licensee and by the staff, both using a 500 cfm unfiltered air inleakage into the control room, are within the dose criterion specified in 10 CFR 50.67. Therefore, the staff finds that the control room habitability assessment performed by the licensee is acceptable.

2.4 Atmospheric Relative Concentration Estimates

2.4.1 Meteorological Data

The licensee made two sets of relative concentration (X/Q) estimates for the control room, EAB, and LPZ dose assessments described above using onsite meteorological data collected during calendar years 1982 through 1986 and 1994 through 1998. These data were measured at 9.6 and 44.9 meters above grade at the Surry site. The licensee confirmed that these data were collected using the guidance in RG 1.23, "Onsite Meteorological Programs." Calibrations were performed at least semi-annually and repairs were made as needed. Data at each level were reviewed every business day by the licensee and compared for consistency and with data from other licensee monitoring sites and National Weather Service data. Monthly statistical reviews were also performed. If needed, staff meteorologists performed further reviews to verify data.

Subsequent to 1996, the licensee set up a program to ensure that the area around the meteorological tower was unobstructed and cleared onsite vegetation in the tower area. However, the licensee found that there were tall trees on privately owned land in an area generally south of the tower. The licensee attempted to resolve this problem with the landowner, but stated that they have been unable to reach a mutually acceptable agreement.

The staff performed a review of the meteorological data using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with

Meteorological Data," on meteorological data quality assurance. Although there were several periods of data outage of at least one parameter in excess of a week during the two 5-year periods, the overall joint data recovery of wind speed, wind direction, and atmospheric stability exceeded 90 percent during each period and yearly recovery was frequently in the high 90 percentiles. Thus, data recovery surpassed the recommended minimum of 90 percent cited in RG 1.23. Further examination of the data indicated that the meteorological measurements were somewhat different than expected when compared to an open site with homogenous flat terrain. This included infrequent intermittent problems in measurement of the temperature lapse rate used to determine stability class, and data indicating A and B stability classes (unstable lapse rates) during the night.

Further, in several cases, the A stability class was reported to occur for more than half a day, some as long as 21 hours. The recorded occurrences were infrequent and, therefore, should have an insignificant impact on the X/Q values calculated for use in this dose assessment. A comparison of the two 5-year periods of data indicated a change in wind direction frequencies between the two periods. When making year-by-year comparisons between the two measurement heights, during 1982 through 1986, lower and upper level wind direction frequency occurrences were closely aligned, whereas during 1994 through 1998, lower and upper level frequency occurrences were somewhat offset from each other. The data analysis also indicated a decrease in wind speeds when comparing the second period with the first period, particularly at the lower measurement level. Thus, the Surry measurements may be affected by temporal and local changes in surface conditions (e.g., tree growth, surface roughness, thermal characteristics). Because of the differences between the two 5-year periods of meteorological data, the licensee made X/Q calculations for each set of data and used the X/Q values that generally resulted in the more limiting consequences in the dose assessments.

2.4.2 EAB and LPZ Relative Concentration Estimates

The licensee recalculated X/Q values for the EAB and LPZ using site-specific inputs and the PAVAN computer code documented in NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Plants." The PAVAN code uses the methodology described in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." The licensee made calculations for the EAB and LPZ for five postulated release locations: the Unit 1 and 2 containment buildings, ventilation vent No. 2, and the east and west louvers of the auxiliary building. All of the EAB distances were modeled as 503 meters. X/Q values for the LPZ were calculated as a function of the postulated release location. In its radiological assessment, the licensee used the X/Q values calculated from 1994 through 1998 data since this resulted in more limiting dose consequences and assumed the Unit 1 containment as the source point of release.

2.4.3 Control Room Relative Concentration Estimates

The licensee used the ARCON96 methodology (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wake") and the 1982 through 1986 meteorological data to calculate control room X/Q values for use in the LOCA, ECCS, RWST and FHA dose assessments. Postulated releases were calculated as ground level point releases other than the containment building X/Q values that were modeled as area sources assuming a factor of six in estimating the initial dispersion coefficients. Wake effects were based on a single containment building and its area above the neighboring auxiliary building. Releases were assumed to travel a straight line distance, not factoring in actual travel over or around the turbine building, and releases from ventilation vent No. 2 were assumed to be released at the height of the top of the vent, assuming no additional rise. The licensee made a number of calculations and found releases from the Unit 1 containment building, ventilation vent No. 2, and personnel airlock to be the limiting cases.

At the Surry Power Station, the control room emergency air intakes draw air from the interior of the turbine building rather than directly from the environment. The licensee made X/Q calculations assuming that air enters the turbine building via the turbine building fresh air intakes or roll-up doors (new receptor locations). These relatively large openings are on sides of the turbine building generally further from the postulated release locations. This results in lower X/Q and dose estimates than assuming that all effluent would enter the turbine building on the side closest to the release locations. The licensee based the selection of these new receptor locations on a recent design change that will result in closure of the turbine building louvers on the side nearest the postulated release locations upon automatic or manual isolation of the control room. The licensee also assumed that there was no further dilution of the effluent by mixing with air in the turbine building as it traveled from the new receptor locations through the turbine building to the control room emergency air intakes. The resultant X/Q values were used in the dose assessment and are listed in Tables 2 and 3.

However, as part of the review, the licensee and staff discussed the possibility of effluent inflow into the turbine building at other locations in addition to the new receptor locations. The licensee performed supplemental calculations assuming that the contribution of all identified pathways, including the new receptor locations, would be on an equivalent basis and only a function of the relative flow area of each pathway. The resultant X/Q values were then compared with the X/Q calculations described in the paragraph above. The licensee noted that the uniform inflow assumptions should be conservative since, for example, actual inflow through the closed louvers should be much lower than that assumed in the uniform flow estimates, that winds blowing effluent toward the louvers should tend to press the louvers more tightly closed, and the X/Q values used in the dose assessment assumed that releases would travel a straight line distance, not factoring in actual travel over or around the turbine building. The licensee's calculations assuming inflow only through the new receptor locations bounded the comparison dose calculations assuming the uniform inflow.

The staff has reviewed the licensee's analysis and performed a confirmatory assessment of the radiological consequence of the postulated LOCA and FHA. The doses calculated by the licensee are listed in Table 1. The doses are all within relevant dose criteria specified in 10 CFR 50.67 and the SRP. Therefore, the staff concludes that the radiological consequences analyzed and submitted by the licensee are acceptable. On the basis of this evaluation, the staff further concludes that the license amendment requested by the licensee is acceptable.

3.0 <u>STATE CONSULTATION</u>

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (66 FR 34289). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

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

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TABLE 1

Radiological Consequences Expressed as TEDE
(rem)

Design Basis Accidents	EAB	LPZ	Control Room[1]
LOCA	24.01	3.57	4.86
Dose criteria[2]	25	25	5.0
Fuel handing accident	6.27	0.27	0.16
Dose criteria	6.3[3]	6.3[3]	5.0[2]

[1] Using 500 cfm unfiltered air inleakage rate
[2] 10 CFR 50.67
[3] SRP 15.0.1

Table 2

Parameters and Assumptions Used in
Radiological Consequence Calculations
Loss-of-Coolant Accident

Parameter	Value
Reactor power	2605 MWt
Containment volume	1.86E+6 ft^3
Sprayed area	1.12E+6 ft^3
Unsprayed area	7.45E+5 ft^3
Containment leak rates	
0 to 1 hour	0.1% per day
1 to 4 hours	0.021% per day
4 hours to 30 days	0
Containment mixing rates	1.49E+6 ft^3/hr
Aerosol removal rates by containment spray (per hour)	
Hours	Rates
0 to 0.027	0
0.027 to 0.060	3.40
0.060 to 0.115	7.92
0.115 to 0.194	1.25
0.194 to 1.140	12.8
1.140 to 1.800	9.47
1.800 to 1.900	6.04
1.900 to 2.020	4.22
2.020 to 2.510	2.25
2.510 to 4.380	1.23
4.380 to 6.480	1.10
6.480 to 720	1.00
Elemental iodine removal rates by spray (per hour)	
Hours	Rates
0 to 0.027	0
0.027 to 2.33	10
2.33 to 720	0

Table 2

Parameters and Assumptions Used in
Radiological Consequence Calculations
Loss-of-Coolant Accident
(Continued)

Containment sump volume	5.83E+4 ft^3
ECCS leak rates	
<u>Hours</u>	<u>Rates</u>
0 to 0.115	0
0.115 to 0.639	1928 cc/hr
0.639 to 720	9600 cc/hr
Iodine partition factor	10%
ECCS leak rates to RWST	
<u>Hours</u>	<u>Rates</u>
0 to 0.639	0
0.639 to 720	200 cc/min
RWST leak rates to environment	
<u>Hours</u>	<u>Rates</u>
0 to 0.639	0
0.639 to 720	1000 cfm
Control room	
Volume	2.23E+5 ft^3
Makeup air flow	1000 cfm
Filter efficiencies	
Aerosol	99%
Elemental iodine	90%
Organic iodine	70%
Unfiltered air inleakage rates	10 and 500 cfm

Table 2

Parameters and Assumptions Used in
Radiological Consequence Calculations
Loss-of-Coolant Accident
(Continued)

Atmospheric Relative Concentrations (X/Q Values)

EAB and LPZ

Exclusion Area Boundary	2 hrs	4.61 E-3 sec/m^3
Low Population Zone	0 - 8 hrs	2.01 E-4 sec/m^3
	8 - 24 hrs	1.22 E-4 sec/m^3
	1 - 4 days	4.18 E-5 sec/m^3
	4 - 30 days	8.94 E-6 sec/m^3

Control Room for LOCA

Unit 1 Containment	0 - 2 hrs	6.74 E-4 sec/m^3
	2 - 8 hrs	5.18 E-4 sec/m^3
	8 - 24 hrs	2.22 E-4 sec/m
	1 - 4 days	1.66 E-4 sec/m^3
	4 - 30 days	1.20 E-4 sec/m^3

(To Turbine Building Fresh Air Intake # 1)

Control Room for ECCS and RWST

Vent Stack No. 2	0 - 2 hrs	6.95 E-4 sec/m^3
	2 - 8 hrs	5.40 E-4 sec/m^3
	8 - 24 hrs	2.30 E-4 sec/m^3
	1 - 4 days	1.71 E-4 sec/m^3
	4 - 30 days	1.22 E-4 sec/m^3

(To Turbine Building Fresh Air Intake # 2 for 0 hr - 4 day time period and to Turbine Roll-up Door # 2 for 4 - 30 day time period.)

Table 3

Parameters and Assumptions
Used in
Radiological Consequence Calculations
Fuel Handling Accident

Parameter	Value
Reactor power	2605 MWt
Radial peaking factor	1.62 (once-burned)
	1.62 (twice-burned)
	1.188 (thrice-burned)
Fission product decay period	100 hours
Number of fuel assembly damaged	1
Fuel pool water depth	23 ft
Fuel gap fission product inventory	
Noble gases excluding Kr-85	5%
Kr-85	10%
I-131	3% (once-burned)
	5.35% (twice-burned)
	6% (thrice-burned)
Other halogens	5%
Alkali metals	12%
Fuel pool decontamination factors	
Elemental iodine	500
Organic iodine	1
Noble gases	1
Atmospheric relative concentrations (χ/Q values in sec/m^3)	
Exclusion area boundary	
0 to 2 hours	4.61E-3
Control Room	
0 - 2 hrs	1.07E-3
(release from Personnel Airlock To Turbine Building Roll-up Door # 2)	
Duration of accident	2 hours

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