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Docket Nos. 50-280
and 50-281

Mr. W. L. Stewart
Vice President - Nuclear Operations
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Stewart:

The Commission has issued the enclosed Amendment No. 84 to Facility Operating License No. DPR-32 and Amendment No. 85 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated September 23, 1982, as supplemented January 17, 1983.

These amendments revise the Technical Specifications to remove a restriction on moving a spent fuel case into the Fuel Building. Requirements are added to install cask impact pads in the spent fuel pool and to not store spent fuel decayed less than 150 days in the first three rows of fuel racks adjacent to the Fuel Building Trolley Load Block.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED

Joseph D. Neighbors, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 84 to DPR-32
2. Amendment No. 85 to DPR-37
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

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No legal objection to Federal Register notice and amendment; SER review not requested.

OFFICE	ORB#1:DL	ORB#1:DL	ORB#1:DL	AD/OR:DL	OELD		
SURNAME	CParrish	DNeighbors	SVarga	GLainas	Cutchin		
DATE	02/16/83	02/18/83:dm	02/18/83	02/18/83	02/28/83		

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 84
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 September 23, 1982, as supplemented January 17, 1983, 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. 84 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 4, 1983

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 85
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 September 23, 1982, as supplemented January 17, 1983, 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. 85 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 4, 1983

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 85 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.10-4	3.10-4
5.4-2	5.4-2
-----	Figure 5.4-1

12. A spent fuel cask or heavy loads exceeding 110 percent of the weight of a fuel assembly (not including fuel handling tool) shall not be moved over spent fuel, and only one spent fuel assembly will be handled at one time over the reactor or the spent fuel pit.

13. A spent fuel cask shall not be moved into the Fuel Building unless the Cask Impact Pads are in place on the bottom of the spent fuel pool.

B. If any one of the specified limiting conditions for refueling is not met, refueling of the reactor shall cease, work shall be initiated to correct the conditions so that the specified limit is met, and no operations which increase the reactivity of the core shall be made.

C. After initial fuel loading and after each core refueling operation and prior to reactor operation at greater than 75% of rated power, the movable incore detector system shall be utilized to verify proper power distribution.

Basis

Detailed instructions, the above specified precautions and the design of the fuel handling equipment, which incorporates built-in interlocks and safety features, provide assurance that an accident, which would result in a hazard to public health and safety, will not occur during refueling operations. When no change is being made in core geometry, one neutron detector is assemblies to assure k_{eff} 0.95, even if unborated water were used to fill the spent fuel storage pit. The spent fuel pool is divided into a two-region storage pool. Region 1 comprises the first three rows of fuel racks (324 storage locations) adjacent to the Fuel Building Trolley Load Block. Region 2 comprises the remainder of the fuel racks in the fuel pool. During spent fuel cask handling, Region 1 is limited to storage of spent fuel assemblies which have decayed at least 150 days after discharge and shall be restricted to those assemblies in the "acceptable" domain of Figure 5.4-1. Administrative controls with written procedures will be employed in the selection and placement of these assemblies. The enrichment of the fuel stored in the spent fuel racks shall not exceed 4.1% weight percent of U-235.

C. Whenever there is spent fuel in the spent fuel pit, the pit shall be filled with borated water at a boron concentration not less than 2,000 ppm to match that used in the reactor cavity and refueling canal during refueling operations.

D. The only drain which can be connected to the spent fuel storage area is that in the reactor cavity. The strict step-by-step procedures used during refueling ensure that the gate valve on the fuel transfer tube which connects the spent fuel storage area with the reactor cavity is closed before draining of the cavity commences. In addition, the procedures require placing the bolted blank flange on the fuel transfer tube as soon as the reactor cavity is drained.

References

FSAR Section 9.5 Fuel Pit Cooling System

FSAR Section 9.12 Fuel Handling System

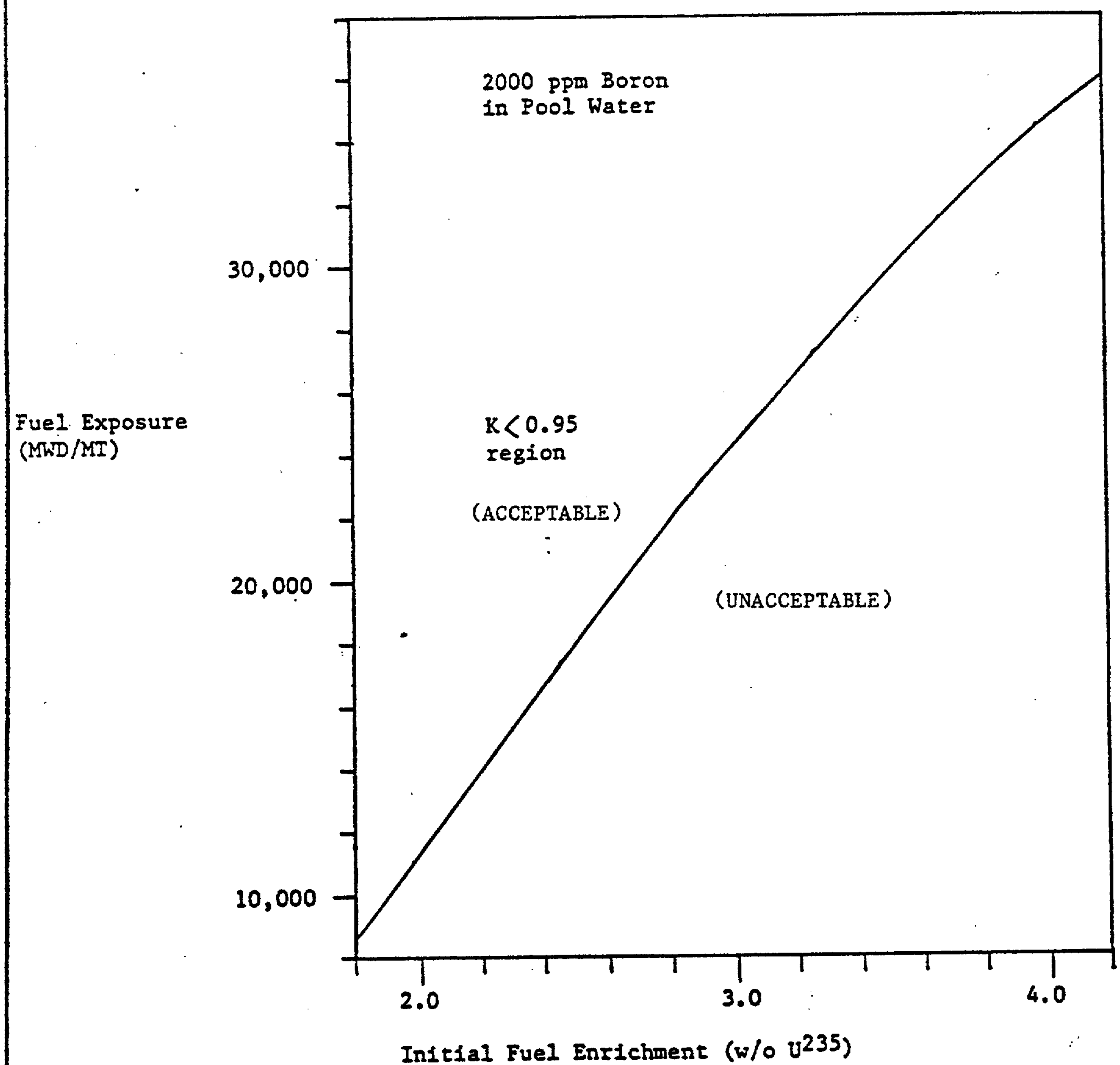

FIGURE 5.4-1

MINIMUM FUEL EXPOSURE VERSUS INITIAL ENRICHMENT
TO PREVENT CRITICALITY IN DAMAGED RACKS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 85 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

INTRODUCTION

By letter dated September 23, 1982, as supplemented January 17, 1983, the Virginia Electric and Power Company (the licensee) requested amendments to the facility operating licenses for Surry Power Station, Unit Nos. 1 and 2. These changes would revise the Technical Specifications related to the movement of spent fuel casks in the fuel building.

DISCUSSION

The existing Technical Specifications prohibit the movement of a spent fuel cask into the fuel building. This restriction exists because the consequence of a drop of a fuel cask into the spent fuel had never been evaluated by the licensee and reviewed by the NRC. The licensee has now provided an analysis of the consequences of dropping a fuel cask in the spent fuel pool and this Safety Evaluation evaluates that analysis.

The licensee has proposed Technical Specifications which would require cask impact pads to be in place on the bottom of the spent fuel pool prior to moving a spent fuel cask into the fuel building. In addition, the licensee has agreed to Technical Specifications which would require that spent fuel assemblies be decayed at least 150 days where stored next to the cask load area.

EVALUATION

The spent fuel pool at Surry Power Station is a rectangular box structure consisting of 6 foot thick reinforced concrete walls and floor resting on piles and soil. The appropriate interior dimensions of the spent fuel pool are about 72.5 feet long by 29.25 feet wide by 39 feet deep. The pool is lined with a 1/4 inch thick stainless steel plate. The fuel area is serviced by a 125-ton capacity trolley for moving spent fuel casks. The cask handling area is at one end of the pool and spent fuel racks are excluded from the area under the trolley rails. The trolley can move only in north-south directions which prevents moving a cask directly over spent fuel. Restraints are provided which prevent displacement of the trolley from the rails during a seismic event.

Our evaluation covered the areas of structural integrity, criticality, radiological consequences, and safe shutdown systems:

Structural

The worst postulated accidents involve dropping of a fully loaded cask into the pool either straight down or at an angle. A cask impact pad is placed in the cask area and, due to the geometry constraints imposed inthe placement of the cask, the cask must hit the impact pad if a drop should occur.

The licensee has conservatively estimated the loads to be expected due to a potential cask drop using a worst-case scenario. Based on this scenario no damage to the pool structure is postulated and perforation of the pool liner is not expected. Even if the liner should be punctured, no significant leakage is expected since the pool walls will not experience through cracking.

Based on our review of the structural analysis performed by the licensee it is concluded that the spent fuel pool will not be damaged by a worst-case cask drop. Accordingly, it was concluded that the proposed cask handling scheme is structurally acceptable and satisfies 10 CFR 50, Appendix A, GDC 2.

Criticality

The criterion for the criticality analysis requires that damage to the fuel storage racks by the postulated heavy load drop does not result in a fuel configuration such that K_{eff} is larger than .95.

The analysis was performed using the Monte-Carlo code KENO-IV to evaluate K_{eff} for the configurations considered. The code essentially solves the Boltzmann transport equation in the distorted geometry of the fuel assemblies after the assumed cask drop. The cross sections were generated using NITAWL and the 123 group XSDRN library. KENO-IV is a well benchmarked code and acceptable to the staff for spent fuel pool calculations. The reported variance for these calculations is less than .005. The following assumptions were used:

- the pool water was borated to 2,000 ppm
- no burnable poisons in the fuel assemblies
- only stable fission products were included, hence, K_{eff} is an upper estimate
- no credit was taken for structural material other than the stainless steel storage cells having a wall thickness of .085 inches.
- 17x17 fuel assembly arrays were assumed. The K_{eff} would be an upper value of 15x15 assemblies were actually present
- spent fuel was assumed stored in the first three rows of assemblies, the remaining racks could contain fuel assemblies having the maximum enrichment allowed.

The analysis sought to determine the maximum values of K_{eff} by varying fuel assembly pitch, initial enrichment, fuel exposure and assembly deformation i.e., crushing of the assemblies.

The licensee has taken adequate administrative measures to divide the storage pool into two parts. One part of the pool consists of the first three rows where spent fuel assemblies can only be stored and the second consists of the remainder of the pool where either spent or fresh assemblies can be located.

We have reviewed the analysis and administrative procedures proposed by the licensee in support of its request to delete the Technical Specification for the Surry Nuclear Power Station which presently prohibits movement of a spent fuel cask into the fuel building without explicit NRC approval.

Based on our review we conclude that the proposed Technical Specification change is acceptable from a criticality standpoint because it meets our requirement of $K_{eff} \leq .95$ for the maximum allowed enrichment of 4.1 w/o U-235.

Radiological Consequences

This section addresses the potential consequences of a postulated cask drop in the Fuel Building to satisfy the evaluation criteria of NUREG-0612, Section 5.1. Also addressed are the potential consequences of postulated cask drops in the Decontamination Building and the truck loading areas.

In accordance with the load handling operation guidelines of NUREG-0612, Section 5.1.2, the licensee has committed to locate the most freshly discharged spent fuel in the spent fuel pool in a location that is separated as much as possible from load transfer paths. For the purpose of this evaluation, therefore, it is assumed that all spent fuel which has not decayed for at least 150 days will not be located in the first three rows of racks in the spent fuel pool. The radiological consequences discussed in this evaluation are contingent upon this minimum 150 day stored spent fuel cooldown time requirement for fuel vulnerable to a cask tip impact. This area of the pool is within 28 feet of the Fuel Building Trolley load block path. The 28 foot separation limit meets the requirements of Alternative 3 to Section 5.1.2 of NUREG-0612, which requires a horizontal minimum separation distance of 25 feet.

The cask loading area is located in the northeast corner of the spent fuel pool. Spent fuel racks are not located in the area vulnerable to a direct cask drop which could damage stored spent fuel; the overhead handling system trolley can move only in a north-south direction, which precludes cask movement over spent fuel storage cells in the pool. Since cask impact pads protect the spent fuel basemat from mechanical damage of any magnitude which could threaten pool integrity, only a cask tip following drop could hypothetically result in the impacting of the nearest three rows of racks of stored spent fuel. For the purpose of this evaluation, a hypothetical cask tip accident is defined whereby a cask falls from an elevation of 1 ft. above the top of the spent fuel pool wall, at an angle, onto a cask impact pad on the pool base. The cask then tips to the west and impacts the adjacent fuel racks. It is assumed that the gap activity of all 324 spent fuel assemblies in the first three rows of racks is released. It is assumed that the fuel has been discharged from the reactor after it has been operating at a steady-state power level of 2546 MW_{th} for an extended period of time.

The licensee's submittal states that their offsite radiological consequence analysis assumes a fuel exposure of 45,000 MWd/MTU; but the information presented in the submittal is not sufficient to perform a consequence analysis for 45,000 MWd/MTU. Specifically, items not addressed are fractional gap activity as a function of burnup and the fuel management scheme, the pool decontamination factor for iodine, and gap gas pressure variations with higher burnup fuel. There is insufficient detail provided for verification of the licensee's analysis. We, therefore, performed independent analysis. Our review was conducted according to the guidance of Standard Review Plans 15.7.4 and 15.7.5, Reg. Guide 1.25, and NUREG-0612 with respect to accident assumptions.

The assumptions in the staff analysis are listed in Table 1 below. The calculated offsite radiological consequences at the Exclusion Area Boundary are 0.42 Rem thyroid and 1 Rem whole body, i.e., small fractions of the dose guidelines of 10 CFR 100.

Table 1: Assumptions in Staff Offsite Radiological Consequence Analysis

Reactor Power Level	2546 MW_{th}
Effective Pool Decontamination Factor for Iodine	100
Radial Power Peaking Factor	1.2
Fuel Exposure for Impacted Spent Fuel Assemblies	25,000 MWd/MTU
Number of Impacted Spent Fuel Assemblies	324

Cooldown Time for Impacted Spent Fuel Assemblies	150 days
Diffusion and Transport Atmospheric Relative Concentration, 0-2 hours At Exclusion Area Boundary	2.1×10^{-3} sec/m^3
No Mitigation Credit for Retention in Fuel Building	

It should be noted that the gap inventory assumptions are based on Regulatory Guide 1.25 assumptions, i.e., assembly average burnup of 25,000 MWd/MTU or less are assumed. Accidents involving fuel with burnup exceeding this value could have increased consequences as a result of potentially higher fission product inventory, higher gap fractions, and lower pool decontamination factors resulting from increased internal fuel rod pressure. We have investigated the potential change in each of these factors and conclude that the change in these factors for fuel burnup levels as high as the currently approved level of 37,000 MWd/MTU batch average at discharge would not be so large as to result in calculated doses in excess of the SRP guideline of 75 rems to the thyroid.

The largest spent fuel casks that can be handled at Surry are the TN-2100 and the GNS-5. These casks are typically designed to hold the equivalent of up to 24 unconsolidated assemblies. The licensee states that spent fuel to be shipped from Surry will have decayed a minimum of two years prior to loading into licensed shipping casks, or five years prior to loading into dry storage casks. Thus, offsite radiological consequences due to cask drop accidents involving release of all gap activity in the fuel assemblies in a cask are bounded by those already calculated in the cask tip analysis in the Fuel Building. This is true because of the fewer number of assemblies involved and the substantial additional decay time of the spent fuel being transported.

Although no safe shutdown equipment is located in the Decontamination Building, there are two potential sources of radioactive releases beneath the cask transfer path. Two liquid waste tanks, the fluid waste treatment tank and the spent resin dewatering tank, are located beneath the transfer path.

We audited the licensee's analysis demonstrating that a cask drop accident resulting in release of the full contents of both tanks to unrestricted areas through groundwater will not result in radionuclide concentrations in unrestricted areas in excess of the maximum permissible concentrations (MPC) of 10 CFR 20, Appendix B, Table II, Column 2. Based upon its review, we agree with the licensee's determination with respect to 10 CFR 20.

We conclude that postulated cask drops impacting 37,000 MWd/MTU batch average spent fuel stored in the spent fuel pool, including decayed fuel at least 150 days old in the first three rows of racks, would result in atmospheric radionuclide releases leading to offsite consequences which are well within the guidelines of 10 CFR 100.

Additionally, we agree with the licensee's determination that a postulated cask drop/liquid spill accident will not result in radionuclide concentrations in an unrestricted area in excess of the maximum permissible concentrations of 10 CFR 20.

Safe Shutdown Capability

There are no safe shutdown systems in the cask travel path. However, the fuel pool cooling system piping in the pool and piping trench at the northeast corner of the pool could be impacted by a cask drop. Damage to this piping will not cause the pool to drain. Temporary repairs can be effected, if necessary, to restore the piping system. A back-up water supply from the fire water hose stations is available for cooling and shielding of the fuel while repairs are made to any damaged piping. This satisfies criterion 4 of NUREG-0612, Section 5.1, and is acceptable.

Based on our review, we conclude that the movement of a full cask into the Fuel Building and the Technical Specifications proposed and agreed upon by the licensee are adequate.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: March 4, 1983

Principal Contributors:
Greg Harrison
Lambros Lois
Millard Wohl
Owen Rothberg

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-280 AND 50-281

VIRGINIA ELECTRIC AND POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 84 to Facility Operating License No. DPR-32 and Amendment No. 85 to Facility Operating License No. DPR-37 issued to Virginia Electric and Power Company (the licensee), which revised Technical Specifications for operation of the Surry Power Station, Unit Nos. 1 and 2, respectively, (the facilities), located in Surry County, Virginia. The amendments are effective as of the date of issuance.

The amendments revise the Technical Specifications to remove a restriction on moving a spent fuel cask into the Fuel Building. Requirements are added to install cask impact pads in the spent fuel pool and to not store spent fuel decayed less than 150 days in the first three rows of fuel racks adjacent to the Fuel Building Trolley Load Block.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since these amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated September 23, 1982, as supplemented January 17, 1983, (2) Amendment Nos. 84 and 85 to License Nos. DPR-32 and DPR-37, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Swem Library, College of William and Mary, Williamsburg, Virginia 23185. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 4th day of March, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing