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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 61
License No. NPF-4

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

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated August 20, 1982 as supplemented by letters dated October 21, 1982, June 16, July 25, September 13, October 28, November 10, December 6, 1983, and April 10, May 8, and May 18, 1984, 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 2.D.(2) of Facility Operating License No. NPF-4 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 61 , 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

James R. Miller

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 21, 1984

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

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a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 9957 ± 10 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 3.4% delta k/k for uncertainties.

b. A nominal 10 9/16 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between new fuel assemblies such that, on a best estimate basis, k_{eff} will not exceed .98, with fuel of the highest anticipated enrichment in place, when aqueous foam moderation is assumed.

5.6.1.3 If new fuel for the first core loading is stored dry in the spent fuel storage racks, the center-to-center distance between the new fuel assemblies will be administratively limited to 28 inches and the k_{eff} shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 288.83 feet. Mean Sea Level, USGS datum.

CAPACITY

5.6.3 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1737 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 45
License No. NPF-7

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

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated August 20, 1982 as supplemented by letters dated October 21, 1982, June 16, July 25, September 13, October 28, November 10, December 6, 1983, and April 10, May 8, and May 18, 1984, 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 2.C.(2) of Facility Operating License No. NPF-7 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 45 , 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

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 21, 1984

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following page of the Appendix "A" Technical Specifications with the enclosed page as indicated. The revised page is identified by amendment number and contains vertical lines indicating the area of change. The corresponding overleaf page is also provided to maintain document completeness.

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5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 3.4% delta k/k for uncertainties.

b. A nominal 10 9/16 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between new fuel assemblies such that, on a best estimate basis, k_{eff} will not exceed .98, with fuel of the highest anticipated enrichment in place, when aqueous foam moderation is assumed.

5.6.1.3 If new fuel for the first core loading is stored dry in the spent fuel storage racks, the center-to-center distance between the new fuel assemblies will be administratively limited to 28 inches and the k_{eff} shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 288.83 feet Mean Sea Level, USGS datum.

CAPACITY

5.6.3 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1737 fuel assemblies.

5.7 COMPONENT CYCLIC or TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

SAFETY EVALUATION

BY THE OFFICES OF

NUCLEAR REACTOR REGULATION AND

NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

RELATED TO INCREASING THE SPENT FUEL STORAGE CAPACITY

AND THE STORAGE OF SURRY SPENT FUEL AT

THE NORTH ANNA POWER STATION, UNITS NO. 1 AND NO. 2

VIRGINIA ELECTRIC AND POWER COMPANY AND

OLD DOMINION ELECTRIC COOPERATIVE

NORTH ANNA POWER STATION, UNITS NO. 1 AND NO. 2

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AND NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

RELATED TO INCREASING THE SPENT FUEL STORAGE CAPACITY

AND THE STORAGE OF SURRY SPENT FUEL AT

THE NORTH ANNA POWER STATION, UNITS NO. 1 AND NO. 2

VIRGINIA ELECTRIC AND POWER COMPANY AND

OLD DOMINION ELECTRIC COOPERATIVE

NORTH ANNA POWER STATION, UNITS NO. 1 AND 2

DOCKET NOS. 50-338 AND 50-339

1.0 Introduction

By letter dated August 20, 1982, the Virginia Electric and Power Company (the licensee) proposed modifications to increase the storage capacity for spent fuel assemblies at the North Anna Power Station, Units No. 1 and 2 (NAPS, NA-1&2). The initial licensed capacity of the spent fuel pool was 400 fuel assemblies, but in 1979 the fuel pool was reracked with high density fuel racks which increased the NAPS storage capacity to 966 assemblies. On August 17, 1979, Amendment No. 14 was issued to Facility Operating License NPF-4 for NA-1 allowing replacement of the fuel racks to accommodate 966 fuel assemblies. The NRC Safety Evaluation for increasing the NAPS storage capacity to 966 assemblies was published as a supporting document to Amendment No. 14 for the NA-1 Operating License NPF-4. The NAPS 966 fuel assembly storage capacity is the configuration presently in place which allows NA-1&2 to operate until 1989 with full core discharge capacity.

The modification proposed by the licensee's August 20, 1982 letter is to replace the presently-installed high density fuel racks with neutron absorber fuel racks which would increase the storage capacity of the spent fuel pool from 966 to 1737 fuel assemblies. The installation of neutron absorber spent fuel racks in the NAPS spent fuel pool would provide full core discharge capability for NA-1&2 until 1997. This storage capacity is based on replacing approximately 33 to 40 percent of the fuel assemblies in the reactor core during each refueling according to the core design parameter and fuel management scheme being utilized. The refueling interval for each unit at NAPS is approximately 18 months. This expanded storage capacity would include the capability to accept a full core discharge from one of the reactor units at any time.

On July 13, 1982, the licensee made application to the NRC for a license amendment for NA-1&2 which would allow the storage of up to 500 spent fuel assemblies from the licensee's Surry Power Station, Units No. 1 and No. 2 (Surry) in the NAPS spent fuel storage area. As early as the spring of 1986, the licensee will lose the ability for full core discharge capability at Surry 1&2. Both Surry Units 1&2 would have to shut down due to the lack of storage for conducting refueling operations in the fall of 1987 and spring of 1988, respectively. Storing 500 Surry spent fuel assemblies in the proposed spent fuel pool storage area at NA-1&2 would provide adequate spent fuel storage capacity (full core discharge capability) for both NA-1&2 and Surry-1&2 through 1992. Inherent in the licensee's amendment request of July 13, 1982, is the necessity for transshipment of 500 Surry spent fuel assemblies from Surry to NAPS.

The proposed neutron absorber spent fuel racks to be installed at NA-1&2 are designed to accommodate either Surry or NAPS fuel assemblies. The spent fuel assemblies for NAPS and Surry are manufactured by the Westinghouse Electric Corporation (W), the nuclear steam system supplier (NSSS) for both power stations. The spent fuel assemblies shipped from Surry to NAPS are 15x15 fuel assemblies. The NAPS fuel assemblies are 17x17 fuel assemblies. The spent fuel is contained in long sealed tubes called fuel rods. A cluster of 204 fuel rods arranged in a 15x15 array make up each of the Surry spent fuel assemblies. Similarly, a cluster of 264 fuel rods arranged in a 17x17 array make up each of the NAPS spent fuel assemblies. A comparison of the physical dimensions of the Surry 15x15 and NAPS 17x17 assemblies is provided in Table 1 below.

Table 1

Comparison of the Physical Dimensions of 15x15 (Surry) and 17x17 (North Anna) Fuel

	15x15 (Surry)	17x17 (North Anna)
Overall Length	159.76	159.8
Overall Dimensions	8.426 x 8.426	8.426 x 8.426
UO_2 Rods Per Assembly	204	264
Guide Tubes Per Assembly	20	24
Number of Grids Per Assembly	7	8
Active Fuel Length	144	144
Cladding Material	Zircaloy - 4	Zircaloy - 4
Clad Thickness	0.0243 (Nominal)	0.0225 (Nominal)

The presently in-place high density racks or the proposed neutron absorber racks to be installed at NAPS are designed to accommodate either Surry or NAPS fuel. In performing the structural/seismic analysis, the thermal hydraulic analysis, and the criticality analysis for the existing and proposed racks, the licensee has used either the Surry or NAPS fuel characteristics that provide the most conservative results. Therefore, the staff's safety evaluation which follows for the licensee's proposed neutron absorber spent fuel racks is applicable to either the storage of Surry or NAPS fuel or a combination of both (maxima of 500 Surry assemblies).

The proposed fuel storage expansion program is limited to the replacement of the current NAPS storage racks with neutron absorber fuel racks. Neutron absorber spent fuel racks permit greater storage capacity by storing the spent fuel assemblies in closer proximity to each other.

The proposed modifications at NAPS will not alter the external physical geometry of the pool or require structural modifications to the fuel building. However, in response to NRC IE Bulletin 80-11, Masonry Wall Design, the licensee has replaced some masonry walls inside the NAPS fuel building. These matters are discussed in Section 2 of this report. Seismic and tornado design provisions as stated in the NA-1&2 Final Safety Analysis Report (FSAR) are not changed as a result of the proposed modification. In addition, these modifications will not affect the leakage and shielding requirements specified in the FSAR. Also, the spent fuel pool cooling and purification system need not be modified to accommodate the proposed increased storage capacity.

Section 2 of this report addresses the radiological aspects of the licensee's proposed modifications to rerack the NA-1&2 spent fuel pool with high density neutron absorber fuel racks for an increased storage capacity of 1737 fuel assemblies.

Section 3 of this report addresses the risk to the health and safety of the public and to transport workers engaged in the proposed transshipment of spent fuel from the Surry reactor site to the NAPS reactor site. This evaluation is concerned only with the actual transportation and does not consider those activities (loading, unloading, etc.) occurring within the protected areas of the reactor sites. Loading and unloading activities within the restricted reactor sites are addressed in the Surry and NAPS FSARs and the staff's Safety Evaluations for Surry and NAPS. Loading and unloading activities associated with the instant proposed modifications at NAPS are addressed in Section 2 of this report. The proposed transshipment evaluation examines the radiological and nonradiological risks that could possibly affect the highway transportation of spent fuel from Surry to NAPS.

Finally, the dates specified in this report for loss of full core discharge and refueling for NAPS and Surry are those stated in the licensee's letter dated May 8, 1984. It is noted that unscheduled shutdowns of the reactor units and/or unknown conditions could impact and stretch out the dates so specified in this report. However, any slippage in these dates (short term) does not mitigate the necessity for the long term actions requested by the licensee.

2.0 Evaluation

2.1 Design Criteria

2.1.1 Description of the Spent Fuel Pool and Racks

The NAPS spent fuel pool serves both reactor units and is located between the reactor buildings with its long axis running East-West. The pool is a concrete box, rectangular in plan view. The walls and floor are approximately six feet thick (thicker in some places) and heavily reinforced. A 3.6 foot thick wall separates the spent fuel area and cask areas. The inside dimensions of the pool are approximately 42.5 feet deep by 57 feet long by 29.25 feet wide, exclusive of the new fuel area and cask area of the East and West ends, respectively. The pool is founded on rock. The top of the pool is at elevation (feet above mean-sea-level) 291.83; the bottom of the pool (inside) is at elevation 249.33; grade is at elevation 271.0. The pool is lined with a continuous one-quarter-inch thick stainless steel liner plate which is anchored to the concrete and is designed for the underwater storage of spent fuel assemblies. The spent fuel pool is designed so that at least 24 feet 1 inch of water is always maintained above the active portions of the spent fuel assemblies stored in the pool. A leak-chase-channel leak detection system is provided.

The existing fuel storage racks are to be replaced with 16 free standing poisoned racks. The capacity of the pool will thereby be increased from 966 spaces to 1737 spaces. The 10 by 12 cell rack is about 10.6 feet long by 8.8 feet wide. All racks are about 14.8 feet high. Racks are primarily constructed of Type 304 stainless steel. Individual square cells are constructed of 0.090 inch thick stainless steel base plate. Each base plate is attached to eight adjustable pedestals.

The proposed spent fuel storage racks are fabricated of Type 304 stainless steel, which is used for all structural components. The individual cells which make up each rack array are in the form of double wall boxes welded to each other with tie plates, so as to maintain a 10 9/16 inch cell pitch. This type of construction provides four compartments which are open to the pool and in which Boraflex neutron absorber elements are placed for criticality control. The Boraflex is positioned on each side of a fuel assembly placed within the cell. The components containing Boraflex are not watertight, thereby eliminating the potential for a pressure buildup within the compartment, for example by radiolysis of entrained water vapor.

2.1.2 Applicable Codes, Standards and Specifications

The proposed racks are designed in accordance with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and revised January 18, 1979 (referred to hereafter as the "NRC Position"). As the basis for structural design of the racks, the NRC Position permits use of Section III, Division 1, Subsection NF of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME code). Buckling of rack components was considered and all buckling design stresses were found to be acceptable when compared with the ASME code as well as the American Iron and Steel Institute Stainless Steel Cold Formed Structural Design Specification (AISI specification).

The existing spent fuel pool was originally designed in accordance with the American Concrete Institute (ACI) Building Code, ACI 318-63. Acceptance criteria and load combinations as set forth in the NA-1&2 FSAR were used in the analysis of the pool structure.

2.1.3 Loads and Load Combinations

Loads and load combinations for the design of the racks were found to be in agreement with the NRC Position. Loads and load combinations for the analysis of the pool structure were found to be in accordance with original NA-1&2 FSAR commitments and are acceptable.

2.1.4 Seismic and Impact Loads

The seismic loads are based on the original design acceleration response spectra calculated for the plant at the licensing stage. This was based on a 0.12 Safe Shutdown Earthquake (SSE) and a 0.06 Operating Basis Earthquake (OBE). Damping values for the racks were taken as 1 percent for OBE and 2 percent for SSE. Impact effects due to fuel bundle/rack interaction as well as rack/pool floor interaction were included in the analysis. Added mass effects of fluid were applied to the racks in a conservative manner.

A separate fuel assembly drop accident analysis was performed. A 2,500 pound object was postulated to impact the top of the rack from a height of 59 inches. The same object was postulated to drop 227 inches through a cell and impact the bottom of the rack. The analysis shows that the dropped object would deform the top of the racks upon impact. However, the analysis further showed that the structural integrity of the racks and the spent fuel pool floor will be maintained.

It was concluded that the seismic and impact loads are in accordance with NRC criteria and are acceptable.

2.1.5 Design and Analysis Procedures

We have reviewed the modeling and analytical procedures used in the seismic analysis of the proposed spent fuel poison storage racks. Also, we have reviewed the methodology used in analyzing the design of the spent fuel structure and liner for the new rack loads. The design and analysis procedures are provided below.

A. Racks

(i) A detailed static analysis model was prepared, from which essential design characteristics were extracted for use in the dynamic analysis models described below.

(ii) A dynamically equivalent response spectrum analysis model was established based on the data generated in step (i). Seismic analysis was then performed using response spectrum analysis methods.

The corresponding inertia forces at each mass point were statically applied to the detailed model created in step (i) and the stress analysis was then performed for various load combinations.

(iii) A separate, non-linear dynamic analysis model was prepared in order to establish the maximum sliding distance for the rack under seismic excitation. A time history analysis was then performed to establish maximum forces acting on the racks under various conditions of friction.

(iv) Tipping and subsequent fall back loads were computed using energy-balance principles. The maximum energy imparted to the rack is established by the analysis in step (ii). The validity of the results in step (ii) were verified by comparing the maximum base shear and base moment resulting from the two separate approaches (steps (ii) and (iii)).

It was found, based on the above analyses, that rack component and weld stresses were within acceptable limits as defined by the more conservative approach of both the ASME code and AISI specification.

B. Pool

The spent fuel pool structure and liner were analyzed for the new rack loads. The criteria used for the original design of the pool, as presented in the FSAR and used for the original and subsequent analyses, were reviewed by the staff.

Thermal loads were included in the original structural design of the pool. Since the temperature of the pool will be maintained at 140 degrees Fahrenheit (°F) for normal conditions and 170 °F for accident conditions as originally designed, no additional thermal analysis of the pool structure was performed. This is acceptable to the staff.

It was concluded that the design of the pool is acceptable for the new rack loads when compared to current criteria for Category I structures. Therefore, the spent fuel pool is satisfactory for the proposed installation. Based on the above, we find that the proposed rack installation will satisfy the requirements of 10 CFR 50 Appendix A, General Design Criteria (GDC) 2, 61 and 62 (as applicable to structures), and is therefore acceptable.

2.1.6 Materials

Rack structural materials are in conformance with the requirements as specified in the ASME code. The spent fuel pool for NA-1&2 is fabricated of materials that will have good compatibility with the borated water chemistry of the spent fuel pool. The corrosion rate of Type 304 stainless steel in this water is sufficiently low to defy our ability to measure it. Since all materials in the pools are stainless steel, no galvanic corrosion effects are anticipated. No instances of corrosion of stainless steel in spent fuel pools containing boric acid has been observed throughout the country.[1] Boraflex has been shown to be resistant to radiation doses in excess of any anticipated in the NA-1&2 spent fuel pool.[2] The venting of the cavities containing the Boraflex to the spent fuel pool environment will ensure that no gaseous buildup will occur in these cavities that might lead to distortion of the racks. We have reviewed the licensee's justification for the lack of a materials surveillance program in the spent fuel storage pool. We conclude that the surveillance programs presently in place with identical material in other existing spent fuel storage pools will provide adequate information on deterioration of materials in these pools. We do not anticipate that such deterioration will occur.

From the above evaluation, we conclude that the corrosion that will occur in the spent fuel storage pool environment should be of little significance during the 40 year life of the plant. Components in the spent fuel storage pool are constructed of alloys which have a low differential galvanic potential between them and have a high resistance to general corrosion, localized corrosion, and galvanic corrosion. Tests under irradiation and at elevated temperatures in borated water indicate that the Boraflex material will not undergo significant degradation during the expected service life of 40 years.

We further conclude that the environmental compatibility and stability of the materials used in the expanded spent fuel storage pool are adequate based on the test data cited above and actual service experience in operating reactors.

We find that the monitoring programs in place with identical material in operating spent fuel storage pools and the selection of appropriate materials of construction by the licensee meet the requirements of 10 CFR Part 50, Appendix A, GDC 61 (having a capability to permit appropriate periodic inspection and testing of components) and GDC 62 (preventing criticality by maintaining structural integrity of components and of the boron poison). Therefore, we find materials compatibility to be adequate and acceptable. We further conclude there will be no significant degradation of materials due to corrosion.

2.2 Criticality

The fuel storage racks consist of double-walled stainless steel tubes having a square cross section with an inner diameter of 8.875 inches, an inner wall thickness of 0.090 inch, a neutron absorber chamber of 0.095 inch width and a cover sheet of 0.029 inch thickness. A Boraflex neutron absorber 0.085 inch thick with a boron-10 loading of .025 grams per square centimeter (gm/cm^2) is placed in the absorber chamber. The storage containers are held on a 10.56 inch center-to-center spacing by the racks.

The nuclear criticality analysis of the racks was performed with the Monte Carlo KENO-IV code with 123 group neutron cross sections prepared with the AMPX-NITAWL code package. This code is widely used for this purpose in the industry and is acceptable. It has been extensively verified and the verification was confirmed for these analyses by comparisons with a series of experiments performed by the Battelle Pacific Northwest Laboratories specially designed to mock up storage rack configurations, including poison curtains between assemblies. The results show that the KENO-IV code conservatively overpredicts the effective multiplication factor (K effective)* by less than one percent. No credit was taken for this overprediction.

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*K effective is the ratio of neutrons from fissions in each generation to the total number lost by absorption and leakage in the preceding generations. To achieve criticality in a finite system, K effective must equal 1.0.

The analysis was performed under the assumption of fresh fuel of 4.3 weight percent U-235 enrichment containing no burnable poison in unborated water. No credit is taken for neutron leakage from the racks and for structural material other than that in the fuel storage containers. Both normal and abnormal configurations were considered. Normal configurations included the reference configuration and those variations in rack dimensions, fuel parameters and fuel location permitted by fabrication tolerances. Fuel and cell parameters used in the analysis are shown in Table 2.2. Abnormal configurations included the results of credible accidents, seismic events, and malfunctions of the fuel pool cooling system.

The results of the analyses showed that the effective multiplication factor for the nominal configuration is 0.935. When all uncertainties (95/95) are statistically combined and added, the result is 0.947. This meets our acceptance criterion (0.95) for this quantity and is acceptable. The effective multiplication factor decreases with increasing water temperature. The results of seismic events are bounded by the eccentric location analysis and are included in the uncertainty band. Placement of assemblies in other than the designated locations is prevented by the structural design. A fuel assembly which lies across the top of the racks is isolated from the fuel in the racks by more than 12 inches of water and is thus neutronically decoupled.

We conclude that any number of fuel assemblies of design similar to the Westinghouse 15x15 or 17x17 design and having an initial enrichment of no greater than 4.3 weight percent U-235 may be safely stored in the proposed North Anna spent fuel racks.

Table 2.2

Fuel and Cell Parameters for Reference Configuration

Westinghouse Fuel Type	17x17	15x15
Fuel Enrichment, w/o	4.3	4.3
Fuel Rod OD, inches	0.374	0.422
Fuel Rod ID, inches	0.329	0.3734
Fuel Rod Pitch, inches	0.496	0.563
Number of Fuel Rods	264	204
Cell Pitch, inches	10 9/16	10 9/16
Cell ID, inches	8 7/8	8 7/8
Cell Wall Thickness, inches	0.090	0.090
Neutron Absorber Material	Boraflex	Boraflex
Neutron Absorber B^{10} Loading, gms/cc^2	0.025	0.025
Neutron Absorber Thickness, inches	0.085	0.085
Neutron Absorber Width, inches	7.5	7.5
Neutron Absorber Length, inches	138	138
Neutron Absorber Chamber Width, inches	0.095	0.095
Cover Sheet Thickness, inches	0.029	0.029

2.3 Thermal Analysis

We have previously found the design for the spent fuel pool cooling and refueling purification system to be acceptable, as discussed in Section 9.1.3 of the NA-1&2 Safety Evaluation Report.

The spent fuel pool cooling portion of the fuel pit cooling and refueling purification system removes residual heat from spent fuel stored in the shared NA-1&2 spent fuel pool. The spent fuel pool cooling system is composed of redundant trains, each train containing a pump and heat exchanger. The redundant trains can be cross-connected so that either pump can provide flow through either or both heat exchangers. The heat exchangers are cooled by component cooling water, with service water available as an emergency supply of cooling water. The spent fuel pool cooling system heat exchangers can be cooled by the component cooling water system associated with either or both reactor units.

The design of the storage pool is such that the fuel will always be covered with water. Because of the locations of fuel pool piping penetrations, the configuration of the pool and the use of siphon breaker vents, no incorrect operation or failure in the fuel pit cooling and refueling purification system could drain the fuel pool water level below elevation 285 feet 9 inches. At this elevation, there is still 24 feet 1 inch of water above the fuel. Makeup water is normally supplied to the spent fuel pool from the boric acid blender in the chemical and volume control system associated with either reactor unit. Assured make up can be supplied from the seismic Category I service water system or the seismic Category I fire protection water system.

The licensee has provided an analysis of the maximum normal and abnormal spent fuel decay heat loads in the spent fuel pool resulting from the proposed increased spent fuel storage capacity. The normal decay heat load results from the maximum number of normal annual refuelings where approximately one third of the fuel asemblies in the core are removed to the spent fuel pool. A full core contains 157 fuel assemblies. All storage spaces in the pool are assumed to be full except for the spaces reserved for a full core offload. For the abnormal decay heat load case, all spent fuel storage spaces are assumed to be full, including a full core offload. For the normal refuelings, the spent fuel is assumed to be removed from the core to the pool instantaneously at 150 hours after shutdown. For the emergency full core offload, the fuel assemblies are assumed to be removed from the core at the rate of 20 minutes per assembly, beginning at 150 hours after shutdown. The licensee calculated a normal decay heat load of 23.1×10^6 British Thermal Units per Hour (Btu/hr) and an abnormal heat load of 39.2×10^6 Btu/hr. These calculated heat loads result in spent fuel pool temperatures of 138.5 °F and 156.5°F, respectively, with one spent fuel pool cooling pump and two heat exchangers in operation. The pool temperature is also calculated to be 135°F (normal heat load) and 151°F (abnormal heat load) if the spent fuel pool cooling system is assumed to be completely operational with both spent fuel pool cooling pumps operating.

The licensee has also provided decay heat load analyses for normal and full-core offload refueling with 500 spaces in the fuel racks occupied by spent fuel transferred from Surry. The presence of Surry fuel increases the decay heat load by a small incremental amount. With 500 Surry fuel assemblies in the NA-1&2 spent fuel pool, the licensee calculates a normal decay heat load of 24.08×10^6 Btu/hr and an abnormal heat load of 40.08×10^6 Btu/hr. These calculated heat loads result in a spent fuel pool temperature of 140°F (normal heat load) and 160°F (abnormal heat load) with one spent fuel pool cooling pump and two heat exchangers in operation. With the spent fuel pool cooling system fully operational, pool temperatures would be reduced to 136°F (normal heat load) and 148.5°F (abnormal heat load).

The licensee has stated that the decay heat load was calculated in accordance with the guidelines of Branch Technical Position (BTP) ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling". An independent calculation by the staff has verified that, in general, the licensee's methodology for calculating spent fuel decay heat is acceptable. However, for the abnormal heat load case, the licensee assumed that the full core offload would take place at the normal refueling time. For the more conservative case with full-core offload occurring 30 days after startup from the previous refueling, the staff calculates a decay heat load of 44×10^6 Btu/hr. This staff value is approximately 10 percent higher than the licensee's calculated value for the abnormal heat load. However, by letter dated April 10, 1984, the licensee verified that the higher heat load would result in an acceptable pool temperature of 152.6°F with the spent fuel pool cooling system fully operational. Thus the spent fuel pool cooling system is capable of maintaining acceptable pool temperatures with the proposed increased spent fuel storage capability.

Based on the above, we conclude that the spent fuel pool cooling system meets the requirements of GDC 44 and the guidelines of BTP ASB 9-2 with regard to decay heat removal capability for the proposed increased spent fuel storage capacity.

2.4 Accident Analysis

2.4.1 Postulated Storage Rack Analysis

The proposed spent fuel storage modifications will provide storage locations for 1737 fuel assemblies and two failed fuel canisters. Each fuel assembly will be stored in a double-walled storage cell of Type 304 stainless steel. The annular spaces between the double walls of the cell contain B_4C (boraflex) neutron absorber elements positioned at the rack height corresponding to the active fuel length of the fuel assemblies. The individual storage cells are welded into rack arrays with array sizes ranging from 9x9 to 11x11 fuel assemblies (9x9, 9x12, 10x11, 10x12, 11x11). This configuration maintains a cell pitch of 10 9/16 inches and prevents placement of a fuel assembly in a location other than a storage cell. As stated previously, the licensee has verified that K_{eff} of the storage fuel configuration is maintained below 0.95 for normal and anticipated abnormal conditions assuming unborated water in the pool and no burnable poison in any of the storage fuel assemblies. Structural and seismic analyses have been performed by the licensee to verify that the rack design is adequate to withstand normal operating, seismic and accident load conditions.

Postulated accidents considered were:

(1) the possibility of the fuel handling bridge fuel hoist grapple becoming hooked on a fuel storage rack, and

(2) The accidental drop of a spent fuel assembly from the highest possible elevation during spent fuel handling onto the storage rack.

For item 1 above, an axial upward force of 4,000 pounds was considered to be exerted on the rack. By letter dated September 13, 1983, the licensee verified that a load limiting device is used to automatically stop upward hoist motion if a preset weight is exceeded. This load limiter setpoint is normally set at 3,400 pounds which conservatively limits any postulated upward force of 4,000 pounds on the storage racks.

For item 2 above (the accidental drop case) a 2,500 pound weight, the maximum load permitted by the NA-1&2 Technical Specifications (TS) over spent fuel, was postulated to drop on the rack from a height of 59 inches above the top of the rack. This postulated 59 inches in drop height is greater than the height at which fuel assemblies are normally raised during fuel handling operations. The results of the seismic and structural analysis indicate that the stresses in the rack structure resulting from the specified load cases are within allowable stress limits for seismic Category I structures. The analysis of the accidental fuel assembly drop condition indicates no buckling or collapse of the storage cells or puncturing of the fuel pool liner (leak tightness integrity of the pool). In all cases analyzed, the value of K_{eff} did not exceed 0.95.

The spent fuel pool and the proposed neutron absorber storage racks are designed to seismic Category I criteria. However, in response to IE Bulletin 80-11, Masonry Wall Design, the licensee advised the staff that 15 of the masonry block walls in the NAPS fuel building do not meet structural design requirements and would be replaced. By letter dated April 10, 1984, the licensee verified that the supports for the replacement walls are designed to seismic Category I requirements. The siding panel assemblies attached to the Category I supports consist of sheet metal, rigid styrofoam and subgirts screwed together as a unit. These assemblies could be dislodged by a design basis earthquake or tornado. However, the licensee verified that the largest panel assembly weighing approximately 650 pounds could not fall into the spent fuel pool with sufficient kinetic energy to result in impact on the spent fuel racks.

Based on the above, we conclude that the spent fuel storage facility and the proposed neutron absorber fuel racks meet the requirements of GDC 2 and GDC 62 with respect to seismic design considerations and prevention of criticality, and the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," Rev. 1, December 1975 and Regulatory Guide 1.29, "Seismic Design Classification," Rev. 3, September 1978 with respect to fuel storage design and design classification and are. therefore, acceptable.

2.4.2 <u>Rack Handling and Installation</u>

The review of heavy load handling at North Anna is being conducted as part of the ongoing generic review initiated by NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants". The results of that review will be reported as part of the Multiplant Action Item C-10. The evaluation provided herein is limited to the heavy load handling activities associated with the proposed spent fuel storage modification.

The replacement neutron absorber spent fuel racks will be offloaded from a truck and brought into the fuel building using the fuel building crane. Movement of the racks inside the fuel building will be done with a special temporary crane which will be inspected and load tested prior to use in accordance with American Nuclear Society Institute Standard (ANSI) B30.2-1976. In addition, the design of the crane will be in accordance with the appropriate guidelines of NUREG-0612.

A special lifting rig, provided by the rack manufacturer, will be used for the positioning and installation of the new racks. The licensee states that the special lifting rig satisfies Guideline 4 of NUREG-0612, Section 5.1.1(4), and will incorporate remotely actuated positive capture devices which will preclude accidentally dropping a rack during handling. In addition, the licensee performed an analysis to verify that the drop of a rack from the highest lift elevation could not result in unacceptable fuel pool damage or loss of spent fuel cooling function. Based on our review, we have verified that the methodology and conclusions of the licensee's analysis are acceptable.

By letter dated September 13, 1983, the licensee provided a description of the rack handling procedures and verified that racks will not be transported above spent fuel. Also, the licensee verified that the procedures used in the rack handling would meet the requirements of Guideline 2 of NUREG-0612, Section 5.1.1(2). Crane operators will be trained in accordance with Guideline 3 of NUREG-0612, Section 5.1.1(3). As discussed in a licensee letter dated April 10, 1984, the operations involved in the assembly and disassembly of the temporary crane will be performed away from the stored spent fuel such that postulated accidents during these operations will not affect spent fuel assemblies.

Based on the above, we conclude that the use of the proposed cranes and load lifting devices for rack handling and installation meets the requirements of GDC 4 and GDC 61 with respect to protection of systems or components important to safety from load drops, and the guidelines of NUREG-0612, Section 5.1.1, with respect to safe load handling practices.

2.4.3 Cask Drop Accidents

The licensee states in a March 22, 1982 submittal addressing NUREG-0612 requirements that the fuel building trolley (1-MH-CR-15) used for moving spent fuel casks does not move over stored fuel. The hook centerline is capable of movement to 5 feet 4 inches from the edge of the fuel pool. The west wall of the fuel pool separates the spent fuel cask storage area from the fuel pool. Only during the movement of spent fuel casks into and out of the fuel building are the casks raised above the top of the fuel pool wall. The centerline of the casks during this movement can be no closer than 1 foot 10 inches from the outside edge of the pool wall. The trolley is equipped with eddy current brakes, dual load holding brakes and "dead man" controls. In addition, the lift height is limited to one foot. These characteristics greatly reduce the likelihood of occurrence of a cask drop, obviating the need for consideration of the radiological consequences of an accident in which a dropped cask would impact stored fuel. Therefore, we conclude that an analysis of the radiological consequences of a cask drop accident is not required.

2.4.4 Spent Fuel Pool Gate Drop Accidents

The licensee states in its March 22, 1982 submittal addressing NUREG-0612 issues that the Fuel Building Movable Platform with Hoists (1-MH-FH-13), which is used to move the fuel cavity gates, is designed to be maneuvered over the spent fuel pool, the fuel transfer canals, and the new fuel handling and storage area as required during fuel handling operations. The movement of the platform is not restricted by electrical interlocks or mechanical stops. Technical Specifications prohibit the movement of loads in excess of 2,500 pounds from travel over irradiated fuel assemblies in the spent fuel pit and the licensee has proposed that plant procedures be revised to prohibit the handling of loads in excess of 2,000 pounds over spent fuel. In addition, administrative procedures will require that the top of the fuel cavity gates be secured to the top of the fuel pool wall by chains during movement of the gates to ensure that the gates, if dropped, will be prevented from tumbling into the fuel pool and damaging the spent fuel racks. Therefore, movement of the spent fuel pool gate should not result in an accident that could result in offsite radiological consequences.

2.4.5 Fuel Handling Accidents

The maximum loads which may be transported over spent fuel in the pool is limited to that of a single assembly. We have previously analyzed the radiological consequences of a fuel handling accident at NA-1&2. The proposed consequences of a fuel handling accident are found in the NA-1&2 staff Safety Evaluation Report of June 1976, wherein the NRC staff stated that the calculated doses are well within the guidelines of 10 CFR Part 100. The proposed modifications for increasing the present storage capacity at NA-1&2 to 1737 fuel assemblies does not increase radiological consequences of fuel handling accidents previously considered in the NA-1&2 Safety Evaluation Report. Therefore, the NRC staff concludes that the radiological consequences of accidents involving fuel handling accidents related to the expansion of storage capacity in the NA-1&2 spent fuel pool meets the acceptance criteria of Standard Review Plan Section 15.7.4 and the guidelines of 10 CFR Part 10C, and therefore, are acceptable.

2.5 Radioactive Waste Treatment

NA-1&2 contain waste treatment systems designed to collect and process the gaseous, liquid, and solid wastes that contain radioactive materials. The waste treatment systems were evaluated in the NRC Safety Evaluation Report dated June 1976, and Supplement No. 2 to the Safety Evaluation Report dated August 1976. There will be no change in the waste treatment systems or in the conclusions given in Sections 9.0 and 11.0 of the evaluation of these systems as a result of the proposed modification. The staffs evaluation of the spent fuel pool cleanup system, in light of the proposed modification, has concluded that any resultant additional burden on the system is minimal because most of the activity will decay away and stop leaking from the spent fuel between refuelings. As a result, the added fuel would contribute little or no additional radioactivity. The existing spent fuel pool cleanup system is adequate for the proposed modification and will maintain concentrations of radioactivity in the pool water to "as low as is reasonably achievable" in accordance with Appendix I to 10 CFR Part 50, and, therefore, is acceptable.

Our evaluation of the radiological considerations supports the conclusion that the proposed modifications to the spent fuel pool at NA-1&2 are acceptable because:

(1) The conclusions of the evaluation of the waste treatment systems, as found in the NA-1&2 Safety Evaluation Report (June 1976) and Supplement No. 2 (August 1976), are unchanged by the proposed modification to the spent fuel pool.

(2) The existing spent fuel pool cleanup system is adequate for the proposed modification.

2.6 Occupational Radiation Exposure

The staff has reviewed the radiation protection related portions of the licensee's plan for the removal and disposal of the presently installed high density fuel racks and the installation of neutron absorber fuel racks. The licensee estimates that the occupational exposure for this operation will be approximately 14.0 person-rems*. This estimate is based on the licensee's breakdown of occupational exposure for each of the following phases of the modification: (1) rack removal/installation, (2) diver operations, (3) fuel shuffle, and (4) disposal of racks. The licensee considered the number of individuals performing a specific job, the average dose rate in the area where the job is being performed, and the worker occupancy time while performing each job.

The spent fuel assemblies themselves contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel (26 feet). The minimum water depth over a fuel assembly while it is being transferred is approximately 9 feet, which results in a dose rate at the water surface of less than 50 mrem/hr.

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*Person-rem is an expression for the summation of whole-body doses to persons in a group. Thus, if each member of a population group of 1,000 people were to receive a dose of 0.001 rem or if two persons were to receive a dose of 0.5 rem each, the total person-rem dose in each case would be 1 person-rem.

One potential source of radioactivity in the spent fuel pool water is radioactive activation of corrosion products (referred to as crud). There are two principal sources of crud in the spent fuel pool water. Crud in the reactor coolant system water mixes with the refueling water and enters the spent fuel pool during refuelings. Also, crud deposits on spent fuel assemblies can be shaken loose in the spent fuel pool during fuel handling. Crud levels in the spent fuel pool are highest during refuelings and decrease continuously over the plant cycle between refuelings. Another source of radioactivity in the spent fuel pool water is fission products. Fission products are released through defects in the spent fuel cladding. Once the fuel is removed from the reactor vessel and is no longer being irradiated, the release rate of fission products from the fuel is greatly reduced. The licensee will use the installed spent fuel pool purification system at NAPS to remove the nonvolatile corrosion and fission product nuclides from the spent fuel pool water. The removal of these nuclides will help to maintain radiation exposure to personnel at low levels. The licensee has estimated that the increased fuel storage at NAPS will have essentially no impact on the concentrations of airborne radioactivity in the fuel building.

The licensee is currently considering two alternative methodologies for disposal of the present spent fuel racks. These are: (1) decontamination of the intact fuel racks; and (2) cutting up of the spent fuel racks for offsite disposal. The licensee will evaluate each alternative method using the following criteria before deciding on which disposal methodology to use:

1. As low as reasonably achievable (ALARA) personnel exposure
2. Minimization of waste volume
3. Cost effectiveness

Based on previous operations for the disposal of spent fuel racks at Surry and.NAPS, the licensee estimates that approximately 1.2 person-rems will result from the disposal of the present spent fuel racks. Rack disposal dose estimates from other utilities have ranged from approximately 0.7 to 5.2 person-rems, depending on the disposal method selected. Dose estimates within this range for other utilities have been previously reviewed and accepted by the staff. The licensee's rack disposal estimate of 1.2 person-rems is within the envelope of this previously accepted range, and is, therefore acceptable. Once the licensee's plans are finalized and available, the licensee is requested to submit for NRC approval the final method for present rack disposal.

In preparation for fuel rack disposal, the licensee will hydrolyze each fuel rack to remove as much contamination as possible prior to removal from the fuel pool. After an element has been lifted out of the water, it will be washed down with demineralized water to remove any remaining loose contamination. When the licensee used similar decontamination techniques at Surry during the 1978 spent fuel pool modification, the resulting fuel rack exposure levels were generally less than 30 mrem with very localized spots having levels greater than 100 mrem

The licensee will use divers to precisely place interface plates, where required, on the spent fuel pool floor and to visually confirm the proper placement of the fuel racks on the fuel pool floor and the interface plates. Following removal of the old fuel racks, the licensee will vacuum the floor of the spent fuel pool to minimize the amount of contamination which would possibly be stirred up by the divers and by the rack installation process. The licensee will position the spent fuel racks currently in the pool for a configuration that will minimize doses to the divers. Health Physics personnel will perform radiation surveys of the areas where the divers will work and will be in constant voice communication with the divers in the pool. The licensee has estimated that the total dose to the diver(s) used for the reracking operation at NAPS will be approximately 4 person-rems.

The staff has reviewed the licensee's spent fuel pool modification report, including a description of the health physics practices which the licensee will implement for the spent fuel pool modification. Many of these practices are similar to those used by the licensee during the similar fuel rack-replacement performed at Surry in 1978. Based on our review of the licensee's report, the staff concludes that the NAPS spent fuel pool modification can be performed in a manner that will ensure that exposures to workers will be as low as is reasonably achievable (ALARA).

The spent fuel pool modification at NAPS will increase the spent fuel pool capacity from 966 fuel assemblies to 1737 fuel assemblies. The spent fuel assemblies (17x17 or 15x15) will in themselves contribute a negligible amount to dose rates in the pool area because of the depth of the water shielding the fuel. The escape of gaseous or volatile fission products from the added spent fuel is expected to be negligible, therefore there will be essentially no impact on concentrations of radioactivity in the fuel building atmosphere. As more fuel is stored in the pool, there is a possibility that increased amounts of radioactive crud will be introduced into the pool. In order to remove these additional impurities, the filters and demineralizer resins will have to be changed on a more frequent basis. The licensee estimates that one additional filter and resin change per year will be required resulting in an additional 0.346 person-rem/year. In 1981, the total personnel exposure from all fuel pool related activities at NAPS was 2.4 person-rems. Based on these figures, the total annual personnel exposure from all fuel pool related activities following reracking should be less than one percent of the total annual occupational radiation exposure at both units. The small increase in radiation exposure due to reracking should not affect the licensee's ability to maintain individual occupational doses at as low as is reasonably achievable levels and within the limits of 10 CFR Part 20. Thus, we conclude that storing additional fuel in the NAPS spent fuel pool will not result in any significant increase in doses received by plant personnel.

There will be personnel exposure associated with the loading and unloading of spent fuel shipping casks used in the transport of Surry spent fuel. The licensee estimates that there will be approximately 150 to 500 cask shipments over an estimated five to six year period. These shipments will result in a total personnel exposure from cask loading/unloading of approximately 28 to 84 person-rems (between 5.6 and 16.8 person-rems per year, based on a five year shipment period). The maximum estimated increase in annual worker dose from cask loading/unloading of 16.8 person-rem (based on 500 shipments over a 5 year period) represents a small fraction (approximately 2.0 percent) of the average annual radiation exposure at NAPS of 816 person-rems*. This

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*The average dose is the average total dose for both units, and is taken from "Occupational Radiation Exposure at Commercial Nuclear Power Reactors 1982," NUREG-0713, Vol. 4, December, 1983.

small increase in radiation exposure due to cask loading/unloading should not affect the licensee's ability to maintain individual occupational doses at as low as is reasonably achievable levels and within the limits of 10 CFR Part 20. Thus, we conclude that loading/unloading activities associated with spent fuel shipping casks will not result in any significant increase in dose received by plant personnel.

2.7 Industrial Security

We have reviewed the proposed modifications to increase the storage capacity for spent fuel assemblies with respect to industrial sabotage. We consider the proposed modifications in the spent fuel pool to have no effect or relevance to the security plan presently in effect for NA-1&2. Our conclusion is based on the fact that the spent fuel pool is designated as a vital area. As a vital area, it is afforded the protection required by 10 CFR Section 73.55 to provide high assurance against successful industrial sabotage by both of the following:

(1) A determined violent external assault, attack by stealth, or deceptive actions, of several persons with tne following attributes, assistance and equipment: (i) well-trained (including military training and skills) and dedicated individuals, (ii) inside assistance which may include a knowledgeable individual who attempts to participate in both a passive role (e.g., provide information) and an active role (e.g., facilitate entrance and exit, disable alarms and communications, participate in violent attack), (iii) suitable weapons, up to and including hand-held

automatic weapons, equipped with silencers and having effective long range accuracy, (iv) hand-carried equipment, including incapacitating agents and explosives for use as tools of entry or otherwise destroying the reactor integrity, and

(2) An internal threat of an insider, including an employee (in any position).

In light of the above, the proposed modifications for reracking with high density fuel racks to increase the spent fuel storage capacity does not change the required level of protection nor the structural design of the external barriers of the pool against the threat of industrial sabotage.

3.0 Transshipment of Surry Spent Fuel to North Anna

3.1 Radiological Assessment During Normal Conditions

This section deals with expected radiation exposures to transport workers and to the public during normal transport conditions and possible radiation risks resulting from transportation accidents.

3.1.1 Radiation Doses To Transport Workers

In its route plan, the licensee has proposed a primary route and four alternate routes designated A, B, C, and D as provided in its July 13, 1982 request for spent fuel shipping route approval. Safeguards route surveys have been conducted by personnel from the NRC Material Transfer Safeguards Licensing Branch on all five routes and, based on the findings of these surveys, the routes have received safeguards approval (see letter from T. S. Sherr, NRC to R. H. Leasburg (licensee) dated July 28, 1982). The licensee commits to having two persons in the transport vehicle through heavily populated areas plus one person each in leading and trailing armed escort vehicles. The radiation dose received by transport workers is estimated as follows.

The dose rate in the cab of the transport vehicle was assumed to be 0.2 mrem/hr (USAEC Report WASH-1238, "Environmental Survey of Transportation of Radioactive Materials To and From Nuclear Power Plants," December 1972, p. 40, Ref. 3). For each route, the amount of each type of road was determined. Vehicle speeds were assumed to be 45 miles per hour (mph) on 6- and 4-lane highways, 35 mph on 2-lane (rural) highways, and 15 mph on city streets. Escort vehicles were assumed to lead and follow, respectively, at a two-second interval from the transport vehicle. The distance between the escort vehicle and the transport vehicle would then be 130 feet, 100 feet, and 45 feet for these three types of highway. The dose rates at these distances were calculated from the equation in Reference 3, p.107. A smaller dose rate factor, K, was used to account for the longer cooling of the Surry fuel as compared with that assumed in Reference 3. For these calculations, a dose rate factor of 390 was used. This value was obtained by multiplying the dose rate factor used in WASH-1238, Reference 3, by the ratios of total dose rates at 10 feet from the cask surface for fuel cooled 730 days and 150 days (See letter dated November 7, 1983, from C. V. Parks, Union Carbide Corporation, Nuclear Division, to R. H. Odegaarden, NRC; Reference 9). For compliance with 10 CFR 71.47(c), the radiation level of 2 meters (approximately 6 feet) must not exceed 10 mrem per hour. To achieve this level, the dose factor must not be greater than 360. The Certificate of Compliance for the TN-8L cask would further restrict the radiation level to 17 mrem per hour at 3 feet. This level corresponds to a dose rate factor of approximately 150. Thus, the dose rate factor used in these calculations is conservative with respect to the requirements specified in 10 CFR 71.47(c).

The total exposure to the two persons in the cab for all 167 shipments would be between 200 and 300 person-mrem, depending on the route. Total exposure to the escorts would be about 40 person-mrem, and the cumulative exposure for all transport workers would be of the order of 300 person-mrem, regardless of the route taken.

The dose to each person in the cab is but a small fraction of the 500 mrem/yr dose restriction for persons in unrestricted areas specified by the National Committee on Radiation Protection and Measurements (NCRP), NCRP-39, 1971 and the International Committee on Radiological Protection (ICRP), No. 2, 1959 and No. 26, 1977.

3.1.2 Radiation Dose to the Public

To calculate the dose received by members of the public, each route was divided into segments for each type of highway and each census unit. Population densities were calculated from 1980 census data and county areas provided in the U. S. Department of Commerce "County and City Data Book", 1977. Based on data as stated in USNRC Report NUREG-0170, "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes", December 1977, p. 4-16, Ref. 4, the traffic densities (vehicles per hour) were estimated to be:

Highway Type	Traffic Density
6-lane Interstate	2800
4-lane Interstate	1480
4-lane Primary	780
2-lane Rural	470

Highway speeds were again assumed to be 45 mph on interstate and 4-lane primary roads, 35 mph on 2-lane rural roads, and 15 mph on city streets. On the average, two persons were assumed to occupy each vehicle on the highway at the same time as a fuel shipment. For all 167 shipments, the cumulative dose to persons on the highway traveling in the same direction as the shipment was calculated using Eq. D-22 (p. D-13 of Reference 4) to be about 200 person-mrem.

From Eq. D-17, (p. D-11 of Reference 4), the cumulative dose for all shipments to persons on the highway traveling in the opposite direction as the shipment was calculated to be a little less than 200 person-mrem. The cumulative dose to all persons living along the shipment route, based on Eq. D-8, (p. D-6, Reference 4), was calculated to be about 40 person-mrem.

The total cumulative exposure to the public for all shipments is the sum of exposures for those persons on the route traveling in the same direction, those traveling in the opposite direction, and those along the route not on the highway. The cumulative exposure would be on the order of 400 person-mrem. This dose would be distributed among many thousands of persons living along the route and traveling on the highways with the shipments. This cumulative exposure is negligible in comparison with the cumulative exposure that this population receives from natural background radiation.

Consider now the maximum exposed individual. Assume a person who lives along the highway and who is within 100 feet of the transport vehicle as it passes on each of its 167 trips. From Reference 3, p. 111, an individual 100 feet from a shipment traveling 8.33 mph would receive an exposure of 0.00058 mrem.

Assuming that the individual postulated in this analysis is in a town where the vehicle speed is estimated to be 15 mph, he would be exposed each trip to (0.00058) x (8.33/15) = 0.0003 mrem per shipment, or a total of 0.05 mrem for all the shipments necessary to ship 500 Surry spent fuel assemblies. This exposure is negligible in comparison with the more than 100 mrem/year of natural background radiation. A person along other portions of the route would receive a smaller dose because the vehicle would pass more quickly than in the example provided above.

3.2.1 Accident Conditions

From Reference 4, accident frequency is about one accident in one million kilometers, or 1.6 accidents per million miles. For a round trip using the longest route (177 miles one way), three vehicles would accumulate 1062 vehicle-miles. (Note that only one vehicle, equivalent to 354 miles, would be transporting the spent fuel cask.) In the 167 round trips to haul 500 assemblies, the total distance would be 59,000 vehicle-miles for the transport vehicle and 177,000 vehicle-miles for all three vehicles combined. The expectation is a little more than one in four (0.25) that one of the vehicles would be involved in a collision sometime during the proposed transshipment and about one in ten (0.1) that the transport vehicle carrying spent fuel would be involved in an accident.

If an accident should occur, it would likely not be serious. An accident severity classification scheme and the relative frequencies of truck accidents for each category are given in Reference 4, indicating that 91 percent of all accidents are in Accident Severity Categories I and II. These may be characterized as "fender benders", and would not be expected to involve the cask nor hinder completion of the shipment. Accidents in Severity Categories III and IV occur 8.6 percent of the time. These accidents would likely result in considerable vehicular damage and possible personnel injury, but would likely not affect the cask or its tie-downs. Categories V, VI, and VII occur in 0.4 percent of all accidents. If an accident of this severity were to occur, one would expect some damage (but not rupture) of the spent fuel cask, major vehicular damage, and serious injury or death of person(s) involved. An accident of Severity Category VIII is expected to occur in only 0.0015 percent of all accidents. The probability of such an accident to the vehicle transporting the spent fuel is considerably less than one in a million (0.000001). A Category VIII accident could conceivably result in a cask rupture.

3.2.2 Vehicle Accident - Cask Not Ruptured

For an accident in which no damage is done to the cask or its tie-downs, the radiological effects would be virtually negligible. An accident involving vehicle overturn, fire, or the cask coming loose from its restraints could present a radiological risk during response and recovery operations, and these risks will be dealt with below.

3.2.3 Risk to Driver and Escort In Transport Vehicle

The most serious radiological risk would occur if the driver and/or escort in the transport vehicle were pinned in the transport cab in proximity to a loaded spent fuel cask. It is expected that the maximum dose rate at one meter from the cask would not exceed 1 rem/hr. Assume also that the time to extricate the person(s) from the wreckage could reach 2 hours. The individual dose received under these circumstances would be 2 rem. This exposure would be far from life-threatening. Furthermore, the physical harm from radiation would likely be much less serious than the bodily injury suffered in the accident.

3.2.4 Risk to Response Personnel

In the accident just postulated, the greatest radiation risk to response personnel would accrue to persons involved in the extrication operations. Although no one person would likely be in the 1 rem/hr radiation field for the entire 2 hours postulated for the extrication, one may consider this exposure as a bounding case. Under these circumstances, each of the rescue persons so engaged would receive a dose of 2 rem, the same as for the driver or escort.

For persons carrying out fire suppression and emergency medical treatment at the scene, one may assume that any one response person spends from 10 to 20 minutes within 10 feet of an accessible surface of the cask. The dose rate at 10 feet from an accessible surface of a loaded cask under the postulated accident conditions would be of the order of 100 mrem/hr. Therefore, the response person might receive a maximum dose of approximately 35 mrem. This dose is small in comparison with the 500 mrem/yr dose restriction for persons in unrestricted areas specified by the National Committee on Radiation Protection and Measurements (NCRR) and the International Committee on Radiological Protection (ICRP).

During the recovery phase, the maximum exposed individual would likely be a rigger fastening chains to the lifting lugs of the cask in order to retrieve it. For brief intervals only, this person may be closer than 3 feet. Assume that the <u>average</u> exposure is equivalent to being at 3 feet for 0.5 hour. The 0.5 hour is considered to be relatively conservative time period since the fastening of chains is a simple procedure. The dose then received by the maximum exposed individual would be 500 mrem. However, the licensee has committed to providing health physics surveillance at the scene, and the rigger (maximum exposed recovery person) would likely be a licensee employee who is under the protection of the licensee's radiation protection program. In that case, a radiation dose standard of 5000 mrem/yr would apply. This 500 mrem maximum exposure is but one-tenth of the radiation standard for occupational exposures.

3.2.5 Risk to Public

Assume that the licensee's Recovery Coordinator who accompanies each shipment will restrain onlookers from approaching the cask at the scene of the accident. Nevertheless, assume a bystander were to remain within 25 feet of an accessible surface of a loaded cask for 0.5 hour. The dose rate at that distance would be approximately 15 mrem/hr. Under those conditions, the bystander would receive a dose of about 7 mrem. This dose is about one-fifth of the radiation received by an individual during a chest X-ray and would not have a detectable health effect.

3.2.6 Cask Ruptured

In the unlikely event of an accident severe enough to rupture a cask containing spent fuel elements, the radiological consequences are described in Sandia National Laboratories Report SAND 83-0867, "A Preliminary Analysis of the Cost and Risk of Transporting Nuclear Waste To Potential Candidate Commercial Repository Sites," June 1983, Reference 7. Reference 7 states "In some recent experiments, contents of a simulated shipping cask for spent fuel were forced out through an opening in the cask... The opening was considerably larger than could result from an accident... However, the amount of material that could be forced out was so small that, if released under the worst possible meterological conditions and in an ultrahigh density urban area, no immediate fatalities would result. Experimental evidence combined with conservative (producing the worst impact) assumptions indicate that only one delayed fatality would result." The population density here assumed was taken from Sandia National Laboratories Report SAND 82-2365, "An Assessment of the Safety of Spent Fuel Transportation in Urban Environs," June 1983, Reference 9, was that for the Manhattan borough of New York City -- a much greater density than that along the proposed transshipment route. Note also that the quoted analysis assumed spent fuel cooled only 150 days, not the 2-year cooling of the Surry fuel prior to transshipment. This minimal consequence, coupled with the extreme improbability of cask rupture, results in a negligible risk from such an occurrence.

4.0 Nonradiological Assessment for Transshipment of Surry Fuel

In this section, those occurrences are considered that would be encountered if the transport vehicle were hauling dogfood, television sets, etc. instead of spent fuel. These dangers include vehicular accidents, pollution from vehicle emmissions, and possible hijacking attempts.

4.1 Risks from Vehicular Accidents

The greatest risk to health and safety presented by the proposed transshipments lies not in the radioactive material being shipped but in the risk of death or injury from traffic accidents. The most recently available analysis of vehicular-accident fatalities and injuries is contained in Reference 7. The shipment by truck of spent fuel from centralized reference points to Hanford, Washington was the basis of these data. The postulated shipments involved a total shipment distance of 320 million kilometers, or 200 million miles (Table 2, p. 9 of Reference 7). For these shipments, 37.3 nonradiological fatalities were predicted from accidents, 8.3 of which were to occupational workers (Table 26, p. 42 of Reference 7).

The longest proposed route from Surry to North Anna is 177 miles, or 354 miles round trip. Assuming three vehicles per shipment, the 167 round trips necessary to complete the transshipment would entail 177,000 vehicle-miles. Using the fatality rates from Reference 7, the expected number of fatalities would be 0.011, of which 0.0024 would be transport workers.

Reference 8 also predicts injuries from accidents. For the proposed transshipments, these statistics would predict 0.14 total injuries, 0.005 of which would be to transport workers. The chances are thus 1 in 200 (0.005) that a transport worker would be injured in a traffic accident.

For verification, these estimates were checked with those calculated from the data given in Reference 3, p. 65. The accident rate cited for trucks carrying hazardous materials was 1.69 accidents per million vehicle-miles. In 177,000 vehicle-miles, one would expect 0.30 accidents. The same source gives fatality and injury rates per accident of 0.039 and 0.51, respectively. These rates would result in 0.011 fatalities and 0.142 injuries. The results of these estimates are:

	Based on data from	
	Ref. 3	Ref. 7
Total vehicular accident fatalities	0.009	0.011
Total vehicular accident injuries	0.15	0.14

The risks from the above estimated vehicular accidents are believed to be acceptably low.

4.2 Risks From Air Pollution

Pollutants emitted by the shipment vehicles can contribute to latent cancer fatalities (LCFs). Data in Reference 7 predict 0.7 LCFs in shipping spent fuel 200 million miles. For the 177,000 miles (longest route) for the transshipments, one would expect 0.00021 LCFs or about one chance in 5,000 of a single death. This risk is negligible when compared with the expected 1 in 5 (0.20) chance of a death from cancer during the life of a member of the public. (Cancer Facts and Figures, American Chemical Society, 1981, p.7.)

4.3 Risks From Increase In Traffic Density

The expected traffic density for various highway types was cited in Section 3.1.2. The increased traffic on the day of a shipment would be less than 0.01 percent for a six-lane interstate highway, where the traffic would be the greatest, and only a fraction of one percent on two-lane rural roads. This small increase is inconsequential when compared with normal fluctuations in traffic flow and does not appear to constitute an unreasonable risk to the safety of the public.

4.4 Risk From A Possible Terrorist Act

The gunplay that might take place in the event of a postulated terrorist attack on a spent fuel shipment could result in the death of one or more drivers, escorts, and persons incidentally in the vicinity. These consequences cannot be minimized. However, so far as is known, the hijacking or sabotage of a spent fuel shipment has never been attempted. Accordingly, the probability of such an attempt cannot be quantified on the basis of historical data.

The following considerations would indicate that the probability of such an event is remote: (1) Extensive safeguards precautions minimize the probability of success. (2) Attempted sabotage, even if successful, would not produce serious radiological consequences (See Section 3.2.6). (3) Attempted theft and separation of plutonium or fission products, even if successful, would require complex equipment and time-consuming reprocessing. (4) The size and weight of the cask and the intense radioactivity of its contents would strongly mitigate against the successful theft of the spent fuel.

On the basis of these considerations, the risk to transportation workers and to the public from a possible terrorist attack is regarded as very small.

5.0 Summary

The staff's evaluation supports the conclusions that the proposed modifications to the spent fuel pool for the North Anna Power Station and the transshipment and storage of the Surry Power Station spent fuel to North Anna is acceptable because:

(1) The structural design and the materials of construction for the spent fuel modifications are adequate and meet the applicable design criteria.

(2) The installation and use of the high density racks will not result in any new means of losing fuel pool integrity or cooling water which has not already been addressed in the licensee's FSAR and the NRC SERs.

(3) The installation and use of the high density racks will not create the possibility of a new or different kind of accident whose consequences would exceed those previously analyzed.

(4) The physical design of the high density racks will preclude criticality for any moderating condition with the limits imposed (for 15x15 and 17x17 fuel assemblies).

(5) The spent fuel pool cooling system has adequate cooling capacity.

(6) The installation and use of the new high density racks can be accomplished safely.

(7) The likelihood of an accident involving heavy loads in the vicinity of the spent fuel pool do not exceed radiological consequences of fuel handling accidents previously evaluated and are well within the guidelines of 10 CFR Part 100 and are acceptable.

(8) The small increase in radiation exposure due to the installation and use of new high density racks should not affect the licensee's ability to maintain individual occupational exposures as low as is reasonably achievable (ALARA) and within the limits of 10 CFR Part 20.

(9) The small increase in radiation exposure due to the storage and loading/unloading of additional spent fuel should not affect the licensee's ability to maintain individual occupational exposures as low as is reasonably achievable (ALARA) and within the limits of 10 CFR Part 20.

(10) The proposed modifications to the spent fuel pool for increasing spent fuel storage does not mitigate the required level of protection from industrial sabotage.

(11) The cumulative exposure to all transport workers engaged in normal transport conditions is well within the 500 mrem/yr dose restrictions for persons in unrestricted areas as specified by the National Committee on Radiation Protection and Measurements and the International Committee on Radiation Protection.

(12) The cumulative exposure to the public for normal transport conditions is negligible in comparison with the cumulative exposure that this population receives from natural background radiation.

(13) The probability of a severe accident to a vehicle transporting spent fuel which could conceivably result in cask rupture is considerably less than one in a million.

(14) The dose rate to the maximum exposed individual engaged in rescue and recovery operations from a severe accident is but one-tenth of the radiation standard for occupational exposures.

(15) The dose rate to the public (bystander) of an accident is postulated to be one-fifth of the radiation received by an individual during a chest X-ray and would not have a detectable health effect.

(16) The risks from estimated vehicular accidents (non-radiological) are believed to be acceptably low.

(17) The risk to transportation workers and the public from a possible terrorist attack is regarded as very small.

6.0 Conclusion

Based on the considerations discussed above, we conclude that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation and transport in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and will not be inimical to the common defense and security or to the health and safety of the public.

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APPENDIX A

CHRONOLOGY OF RADIOLOGICAL REVIEW

REGARDING

SPENT FUEL EXPANSION

AND STORAGE OF SURRY POWER STATION

SPENT FUEL AT THE NORTH ANNA POWER STATION

NOTE: Documents referenced in this chronology are available for public inspection and copying for a fee at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Public Document Rooms located at Board of Supervisors Office, Louisa County Courthouse, Louisa, Virginia 23093 and the Alderman Library, Manuscripts Department, University of Virginia, Charlottesville, Virginia 22901

July 13, 1982	Letter from R. H. Leasburg (licensee) to H. R. Denton, NRC, for a license amendment to permit the storage of 500 Surry spent fuel assemblies at NAPS.
July 13, 1982	Letter from R. H. Leasburg (licensee) to Robert F. Bernett (sic).

July 28, 1982	Letter from Theodore S. Sherr, NRC, to R. H. Leasburg (licensee).
August 20, 1982	Letter from R. H. Leasburg (licensee) to H. R. Denton, NRC, for a license amendment to modify spent fuel storage to 1737 fuel assemblies.
October 21, 1982	Letter from R. H. Leasburg (licensee) to H. R. Denton, NRC, transmitting additional information on spent fuel pool heat loads.
June 16, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, transmitting responses to NRC questions.
July 19, 1983	Letter from R. Clark, NRC, to W. L. Stewart (licensee) requesting information regarding the transshipment of fuel from Surry to North Anna.
July 25, 1983	Letter from R. Clark, NRC, to W. L. Stewart (licensee) requesting additional information regarding spent fuel storage expansion.
September 13, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, transmitting responses to NRC request for additional information.

October 28, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, transmitting responses to NRC request for additional information.
November 10, 1983	Letter from J. R. Miller, NRC, to W. L. Stewart requesting additional information regarding spent fuel storage capacity.
November 23, 1983	Letter from W. L. Stewart to H. R. Denton, NRC, advising when NRC request for additional information will be provided by licensee.
December 6, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing additional information regarding spent fuel capacity expansion.
December 6, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing additional information regarding increase in spent fuel capacity.
December 14, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, correcting typographical errors in the licensee letter dated December 6, 1983.
December 14, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing clarification for neutron absorber spent fuel rack.

December 29, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing additional information regarding spent fuel capacity expansion.
December 29, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing additional information of proprietary nature regarding neutron absorber racks.
April 10, 1984	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing additional information regarding spent fuel expansion.
May 8, 1984	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing additional information regarding available spent fuel storage.
May 18, 1984	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing additional information regarding low level radwaste.

APPENDIX B

REFERENCES

(1) J. R. Weeks, "Corrosion of Materials in Spent Fuel Storage Pools," BNL-NUREG-23021, July 1977.

(2) J. S. Anderson, Brand Industries, Inc., Reports 748-2-1 (August 1978), 748-10-1 (July 1979), 748-30-1 (August 1979).

(3) USAEC Report WASH-1238, "Environmental Survey of Transportation of Radioactive Materials To and From Nuclear Power Plants," December 1972.

(4) USNRC Report NUREG-0170, "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes", December 1977.

(5) Rand-McNally Road Atlas, 1982.

(6) U. S. Department of Commerce, "County and City Data Book", 1977.

(7) Edwin L. Wilmot, Marcella M. Madsen, Jonathan W. Cashwell, and Davis S. Joy, Sandia National Laboratories Report SAND83-0867, "A Preliminary Analysis of the Cost and Risk of Transporting Nuclear Waste To Potential Candidate Commercial Repository Sites", June 1983.

(8) R. P. Sandoval, J. P. Weber, H. S. Levine, A. D. Romig, J. D. Johnson, R. E. Luna, G. J. Newton, B. A. Wong, R. W. Marshal, Jr., J. L. Alvarez, and F. Gelbard, Sandia National Laboratories Report SAND82-2365, "An Assessment of the Safety of Spent Fuel Transportation in Urban Environs," June 1983.

(9) Letter dated November 7, 1983, from C. V. Parks, Union Carbide Corporation, Nuclear Division, to R. H. Odegaarden, U.S. N.R.C.

ENVIRONMENTAL ASSESSMENT

BY THE OFFICES OF NUCLEAR REACTOR REGULATION

AND NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

RELATED TO INCREASING THE SPENT FUEL STORAGE CAPICITY

AND THE STORAGE OF SURRY SPENT FUEL AT

THE NORTH ANNA POWER STATION, UNITS NO. 1 AND NO. 2

VIRGINIA ELECTRIC AND POWER COMPANY AND

OLD DOMINION ELECTRIC COOPERATIVE

NORTH ANNA POWER STATION, UNITS NO. 1 AND 2

DOCKET NOS. 50-338 AND 50-339

JULY 2, 1984

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NORTH ANNA POWER STATION, UNITS NO. 1 AND 2
DOCKET NOS. 50-338 AND 50-339

1.0 Introduction

The spent fuel storage capacity of the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2, NAPS), was 400 spent fuel assemblies when NA-1 was licensed in 1978. NAPS spent fuel is stored in a spent fuel pool common to both NA-1&2. This licensed capacity was increased in 1979 to 966 fuel assemblies by reracking the spent fuel pool with high density racks. The spent fuel storage capacity at the Surry Power Station, Units No. 1 and 2 (Surry 1&2, Surry) was 464 spent fuel assemblies when Surry 1 was licensed in 1972. This licensed capacity was increased in 1979 to 1,044 spent fuel assemblies. This limited increase in storage capacity at Surry and NAPS was in keeping with the expectation generally held in the industry that commercial fuel processing would not provide near-term relief from diminishing available storage locations.

Commercial reprocessing of spent fuel has not developed as had been originally anticipated. In 1975 the Nuclear Regulatory Commission directed the staff to prepare a Generic Environmental Impact Statement (GEIS, the Statement) on spent fuel storage. The Commission directed the staff to analyze alternatives for the handling and storage of spent light water power reactor fuel with particular emphasis on developing long range policy. The Statement was to consider alter-native methods of spent fuel storage as well as the possible restriction or termination of the generation of spent fuel through nuclear power plant shutdown.

A Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel (NUREG-0575), Volumes 1-3 (the FGEIS) was issued by the NRC in August 1979. In the FGEIS, consistent with long range policy, the storage of spent fuel is considered tc be interim storage, to be used until such time that the issue of permanent disposal is resolved and implemented.

One spent fuel storage alternative considered in detail in the FGEIS is the expansion of onsite fuel storage capacity by modification of the existing spent fuel pools. Since the issuance of the FGEIS, applications for approximately 113 spent fuel pool capacity expansions have been received and 102 have been approved. The remaining 11 are still under review. The finding in each case has been that the environmental impact of such increased storage capacity is negligible. However, since there are variations in storage designs and limitations caused by the spent fuel already stored in some of the pools, the FGEIS recommends that licensing reviews be done on a case-by-case basis to resolve plant specific concerns.

In addition to the alternative of increasing the storage capacity of the existing spent fuel pools, the FGEIS discusses in detail other spent fuel storage alternatives. The finding of the FGEIS is that the environmental impact costs of interim storage are essentially negligible, regardless of where such spent fuel is stored. A comparison of the impact-costs of various alternatives reflects the advantage of continued generation of nuclear power versus its replacement by coal fired power generation. In the bounding case considered in the FGEIS, that of shutting down the reactor when the existing spent fuel storage capacity is filled, the cost of replacing nuclear stations before the end of their normal lifetime makes this alternative uneconomical.

This Environmental Assessment (EA) addresses only the specific environmental concerns related to the proposed expansion of the NA-1&2 spent fuel storage capacity and the storage of 500 Surry spent fuel assemblies at NA-1&2. This EA consists of four major parts, plus a summary and conclusion. The four parts are: (1) descriptive material, (2) an appraisal of the environmental impact of the proposed actions, (3) an appraisal of the environmental impact of postulated accidents, and (4) the enviromental impact of the proposed transshipment of spent fuel from Surry to NAPS.

Pursuant to the provisions of 10 CFR 51.30, the need for the proposed actions is specified in Section 1.2 of this EA. The alternatives and impacts for the proposed actions are described in Section 1 of this EA and in the FGEIS. The environmental impacts for the proposed actions are provided in Section 3 through Section 6 of this EA. No other agencies or persons were consulted in the NRC staff's preparation of this EA. Finally, the identification of sources used in preparing this EA is provided in Appendices A and B.

1.1 Description of the Proposed Action

By application dated July 13, 1982, the Virginia Electric and Power Company (the licensee) proposed an amendment to the NA-1&2 Facility Operating Licenses Nos. NPF-4 and NPF-7 which would allow the licensee to receive from Surry 1 & 2, possess, and store in the NA-1&2 spent fuel pool irradiated Surry (spent) fuel assemblies containing special nuclear material, enriched to not more than 4.1 percent by weight U-235. Inherent in the licensee's above proposed action is the required transshipment of the Surry spent fuel from Surry to NAPS.

By application dated August 20, 1982, the licensee proposed an additional amendment to the NA-1&2 Facility Operating Licenses which would allow the installation of neutron absorber spent fuel storage racks at NA-1&2 which would increase the spent fuel storage capacity from the present 966 assemblies to 1737 assemblies.

The environmental impacts associated with NA-1&2 were considered in the NRC's Final Environmental Statement (FES) dated April 1973. An Environmental Impact Appraisal (EIA) for increasing the NA-1&2 spent fuel storage capacity from 400 to 966 fuel assemblies was completed on April 2, 1979 and published as part of Amendment No. 14 (August 17, 1979) to the NA-1 Facility Operating License No. NPF-4. The purpose of this EA is to evaluate any additional environmental impacts which are attributable to the proposed increase in the spent fuel pool storage capacity and storage of Surry fuel at NA-1&2.

1.2 Need for Increased Storage Capacity

The basic reason for the licensee's proposed actions is to prevent both loss of full core discharge and loss of refueling capability at the Surry and North Anna Nuclear Power Stations.

Whenever the licensee refuels Surry 1 or 2 and NA 1 or 2 (replacing 33 to 40 percent of the fuel assemblies in the reactor core) it must have room to store the spent fuel that is removed from the reactor. For each reactor, these refuelings occur at intervals of approximately every 18 months. In addition, the licensee believes it must maintain the ability to discharge the full core in a particular reactor at any time. This "full core discharge capability" is essential whenever inspections or repairs necessary for continued operations require the offloading of the entire core from the reactor.

There are presently 769 spent fuel assemblies being stored in the Surry spent fuel pool. As early as the spring of 1986, the licensee will lose the ability to remove all of the fuel from either of its reactors at Surry. Full-core discharge capability has been required three times in the past to perform necessary maintenance or repairs at Surry, and will most likely be required in the future. In 1979, all fuel had to be removed from Surry 2 and stored in the spent fuel pool so that the unit's steam generators could be replaced. The fuel from Surry 1 had to be stored in the spent fuel pool in 1980 while the same work (replacement of Surry 1 steam generators) was performed. During the outage of Surry 2 in late 1981, full-core discharge was necessary to complete maintenance on the unit's residual heat removal system. Full-core discharge was necessary again during the refueling outage of Surry 2 in the late spring of 1983, to perform required in-service inspection of the unit's reactor vessel. Full-core discharge was also necessary at NA-1 during the spring through early winter of 1982 to replace control rod guide tube assemblies. Both Surry 1 & 2 would have to be shut down in the fall of 1987 and spring of 1988, respectively, due to the lack of storage space for conducting refueling operations. In evaluating its Surry facility, the licensee has found that no additional fuel over its present licensed capacity may be stored in the Surry spent fuel pool without exceeding structural design criteria. These matters, therefore, increase the need for additional spent fuel storage.

The NA-1&2 spent fuel pool presently has a storage capacity of 966 spent fuel assemblies. Two hundred and ninety-three (293) spent fuel assemblies are presently stored in the pool. Without any storage of Surry fuel and any increase in storage capacity, the present NA-1&2 spent fuel pool will lose full core discharge capability in 1989, and NA-1&2 would have to be shut down in 1991 and 1990, respectively due to lack of storage for refueling operations. Therefore the licensee has proposed to increase the spent fuel storage capacity at NA-1&2 from 966 storage locations to 1737 storage locations through the use of neutron absorber racks.

If only the proposed neutron absorber racks should be installed at NAPS, NA-1&2 would not lose full core discharge capability until 1997, and the two units would not be required to shut down until 2000 and 1999, respectively. If the proposed neutron absorber racks are installed and 500 Surry spent fuel assemblies are shipped and stored at NAPS, NA-1&2 would not lose full core discharge capability until 1992, and the two units would not be required to shut down until 1994 and 1993, respectively. The storage of 500 Surry assemblies at NAPS would extend the loss of full core discharge at Surry-1&2 from 1986 to 1992 and would postpone the shutdown dates for Surry-1&2 from 1987 and 1988 to 1993 and 1994, respectively.

If neutron absorber spent fuel racks were for some reason not installed at NA-1&2, the number of Surry assemblies to be shipped to NA-1&2 would be decreased to approximately 150. Storage of 150 Surry spent fuel assemblies at

NA-1&2 would extend the loss of full core discharge capability for Surry until fall 1987, and would postpone the shutdown dates for the Surry-1&2 until 1990 and 1989 respectively. Similarly, storage of 150 Surry spent fuel assemblies at NA-1&2 would extend the loss of full core discharge at NA-1&2 until 1987 and shutdown dates would be 1990 and 1989, respectively.

Based on the above, to avoid future unit shutdowns due to lack of spent fuel storage space and given the uncertainty of fuel reprocessing or a permanent solution to the spent fuel problem, the licensee's proposed actions specified in its July 13 and August 20, 1982 applications are timely and justified.

1.3 Fuel Reprocessing History

Currenty, spent fuel is not being reprocessed on a commercial basis in the United States. The Nuclear Fuel Services (NFS) plant at West Valley, New York was shut down in 1972 for alterations and expansion; in September 1976, NFS informed the Commission that it was withdrawing from the nuclear fuel reprocessing business. The Allied General Nuclear Services (AGNS) proposed plant in Barnwell, South Carolina, is not licensed to operate.

The General Electric Company's (GE) Morris Operation (MO) in Morris, Illinois is in a decommissioned condition. Although no plants are licensed for reprocessing fuel, the storage pool at Morris, Illinois and the storage pool at West Valley, New York are licensed to store spent fuel. The storage pool at West Valley is not full, but NFS is presently not accepting any additional spent fuel for storage, even from those power generating facilities that had contractual arrangements with NFS. On May 4, 1982, the license held by GE for spent fuel storage activities at its Morris operation was renewed for another 20 years; however, GE is also not accepting any additional spent fuel for storage at this facility, except where previous contractual obligations may require it to do so.

2.0 Facility

The principal features of the spent fuel storage and handling at NA-1&2 as they relate to the proposed modification are described below to aid in understanding the evaluations provided in subsequent sections of this EA.

2.1 Spent Fuel Pool (SFP)

Spent fuel assemblies, (when initially removed from the core), are intensely radioactive due to their fresh fission product content and they have a high thermal output. The SFP was designed for storage of these assemblies to allow for radioactive and thermal decay prior to shipping them to a reprocessing facility. The major portion of decay occurs in the first 150 days following removal from the reactor core. After this period, the spent fuel assemblies may be withdrawn and placed in heavily shielded casks for shipment. Space permitting, the assemblies may be stored for longer periods, allowing continued fission product decay and thermal cooling.

The spent fuel pool for NA-1&2 is common to both units. The pool is a concrete box, rectangular in plan view. The walls and floor are approximately six feet thick and heavily reinforced. Inside dimensions of the pool are approximately 42 feet deep by 57 feet long by 29 feet wide. The pool is founded on bed rock. The pool is lined with a continuous one-quarter inch thick stainless steel liner plate which is anchored to the concrete and is designed for the underwater storage of spent fuel assemblies. The spent fuel pool is so designed that at least 24 feet 1 inch of water is always maintained above the active portions of the spent fuel assemblies stored in the pool. The liner plate provides leak tight integrity for the spent fuel pool.

2.2 Spent Fuel Pool Cooling and Purification System

The spent fuel pool is provided with a cooling system to remove residual heat from the fuel stored in the pool. Purification equipment is provided to maintain the quality and clarity of the water in which the fuel assemblies are immersed. This system is discussed in Section 9.1.3 of the NA-1&2 Safety Evaluation Report (SER).

The cooling system is designed to maintain the pool water temperature at or below 140°F under normal refueling conditions. Two cooling loops are provided, each with a full capacity (2750 gpm) circulating pump and a heat exchanger designed to remove heat from the pool at a rate of 56.8×10^6 British Thermal Units per hour (BTU)/hr. The two loops are also cross-connected for flexibility in the event of a component failure.

In operation, a circulating pump draws water from one end of the pool, circulates it through a heat exchanger and returns it to the other end of the pool. Purity of the water is maintained by passing a portion of the water, approximately 130 gpm, through a 45 cubic feet (ft^3) demineralizer and filter. Three purification pumps, two filters and one demineralizer are provided for this function. There is also a skimmer system to remove surface dust and debris from the spent fuel pool. Based on the present design and capacity of these systems, no changes are required due to the proposed spent fuel modifications.

2.3 Radioactive Waste Treatment System

The station contains waste treatment systems designed to collect and process the gaseous, liquid and solid wastes that might contain radioactive material. These waste treatment systems for NA-1&2 are evaluated in the FES dated April 1973 and the Addendum to the FES dated November 1976. No changes in these systems are required due to the proposed spent fuel pool modifications.

3.0 Non-Radiological Environmental Impacts of Proposed Actions

We have reviewed the material submitted by the licensee in support of the proposed amendment applications. Our review of the nonradiological environmental impacts resulting from the replacement of the fuel racks is discussed below.

The original spent fuel pool design for NA-1&2 provided space for 400 fuel assemblies. In 1979 a license amendment was issued to the licensee by the NRC allowing replacement of the fuel racks to accommodate 966 fuel assemblies. An EIA dated April 2, 1979 of that fuel pool modification was performed and published as part of Amendment No. 14 to the NA-1 Facility Operating License No. NPF-4 on August 17, 1979. This current review is thus the second review for increasing the spent fuel capacity at NA-1&2.

The increase in fuel pool capacity at NA-1&2 is achieved by removing the existing fuel racks which hold fuel assemblies at a center-to-center spacing of 14 inches and replacing them with new racks which have a center-to-center spacing of 10 9/16 inches. There is no structural modification to the fuel pool. The structural members of the new fuel racks are Type 304 stainless steel. Included in the racks are Boraflex neutron absorber elements positioned at either side of each assembly fuel region. The new racks are assembled off site. The new assemblies are brought in by truck and unloaded right at the fuel pool building.

As was found with the April 2, 1979 review, there is no new commitment of land resources, and no on site construction involved. In addition, the type of use will remain unchanged by the proposed modifications. The additional storage capacity to be provided by the proposed modifications would result in more efficient use of the land already designated for NA-1&2 spent fuel storage. Therefore, any impact to terrestrial resources is insignificant.

With the pool filled to capacity, heat will be generated at a greater rate with the new racks than with the existing racks. The existing racks would accept all NA-1&2 spent fuel through the year 1989 without the loss of full core discharge capacity. The oldest fuel in the pool at that time would be out of the reactor for about 10 years. With the new racks, this older fuel would be left in the pool as newer spent fuel is added. With part of the pool allocated to storage of Surry fuel, the expanded pool would accommodate spent fuel from NA-1&2 for another four and five years, respectively.

The proposed modifications would store 500 Surry spent fuel assemblies in the NA-1&2 spent fuel pool. These 500 assemblies would have been removed from the Surry reactors for no less than two years prior to shipment and could have been cooled as long as 10 years. These assemblies would be brought to NAPS over a five to six year period starting in 1985. Thus when the NA-1&2 spent fuel pool would be filled to capacity in 1994 (1,737 assemblies), it would contain 500 Surry assemblies with a minimum out-of-reactor age of 8 to 11 years. These older Surry assemblies will contribute only a small fraction to the total heat generation in the filled NA-1&2 spent fuel pool.

The expected rate of decay heat generation with the present spent fuel pool filled with 966 assemblies is 19.4×10^6 Btu/hr. Under full core discharge conditions the expected heat generation would be 35.9×10^6 Btu/hr. With the pool modified to accommodate 1,737 fuel assemblies and 500 of those being from Surry, the heat rate would increase to 23.1×10^6 Btu/hr. Under full core discharge conditions, the heat rate would increase to 40.1×10^6 Btu/hr. Thus the retention of the older assemblies for the proposed modifications adds approximately 15 percent to the spent fuel pool heat load.

This waste heat is transferred by the closed loop Spent Fuel Pool Cooling System to the component cooling water system. This closed system transfers the heat to the auxiliary cooling system. The auxiliary cooling system discharges the heat to the service water reservoir where most of the heat is transferred to the atmosphere by spray cooling. The dominant source of waste heat from the station is the condenser cooling water. The average rate of heat discharge from NA-1&2 is 13.5×10^9 Btu/hr (FES, page 3-17). The heat from the spent fuel pool is about one tenth of one percent of this amount. The increase in the fuel pool heat discharge because of retaining the older spent fuel assemblies is about two one-hundredths of one percent of the total NA-1&2 heat discharge. This is insignificant in relation to total station discharge.

The additional heat would increase evaporation from Lake Anna by about 8 gallons per minute if all of the heat were transferred to the atmosphere by evaporation. This is small in comparison to total station water use which is about a million gallons per minute.

No change is necessary in the Fuel Pool Purification System to accommodate additional spent fuel assemblies. Since most of the fuel pool contamination occurs during fuel transfer and since the number and frequency of refueling operations will not change, there will not be a significant increase on the system due to the increased storage capacity. There is no direct discharge from the pool to other water systems. There will be no change in usage or discharge of chemicals from the station. Thus there will be no water quality impact different from that previously reviewed.

As discussed in our evaluation above, we find:

(1) The proposed modifications will alter only the spent fuel storage racks. It will not alter the external physical geometry of the spent fuel pool structures. In addition, construction of the new racks will be done offsite and transported to the facility. No unusual terrestrial effects are anticipated or considered likely.

(2) Additional storage will not result in a measurable increase in non-radiological chemical waste discharges to the receiving water. The licensee does not propose any change in chemical usage or change to the NPDES permit.

(3) Additional storage will not result in measurable thermal effects to the receiving water. The increase in the heat load due to this modification is about two one-hundredths of one percent of the total NA-1&2 station discharge and is insignificant.

Therefore, we conclude, based on the above, that the spent fuel pool modifications will not result in non-radiological environmental effects significantly greater or different from those already reviewed and analyzed in the FES for NA-1&2.

4.0 Radiological Environmental Impacts of Proposed Action

4.1 Introduction

The potential radiological environmental impacts associated with the expansion of the spent fuel storage capacity have been evaluated and are addressed below.

During the storage of the spent fuel under water, both volatile and non-volatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54 which are not volatile. The radionuclides that might be released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90, are also predominantly nonvolatile. The primary impact of such nonvolatile radioactive nuclides is their contribution to radiation levels to which workers in or near the spent fuel pool would be exposed. The volatile fission product nuclides of most concern that might be released through defects in the fuel cladding are the noble gases (xenon and krypton), tritium and the iodine isotopes.

Experience indicates, however, that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the spent fuel water appear to be radionuclides that were present in the reactor coolant system prior to refueling (which become mixed with water in the spent fuel pool during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the spent fuel pool.

During and after refueling, the spent fuel purification system reduces the radioactivity concentrations to low levels. It is theorized that most failed fuel contains small, pinhole-like perforations in the fuel cladding at the reactor operating condition of approximately 800°F. A few weeks after refueling, the spent fuel is cooled in the spent fuel pool and the fuel clad temperature becomes relatively cool, approximately 180°F. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the gap between pellets and clad, thereby tending to retain the fission products within the gap. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months. Based on the operational reports submitted by the licensees and discussions with the operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the MO (formerly Midwest Recovery Plant) at Morris, Illinois, or at the Nuclear Fuel Services (NFS) storage pool at West Valley, New York. Some spent fuel assemblies which had significant leakage while in operating reactors have since been stored in these two pools. After storage in the onsite SFP, these fuel assemblies were later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from these fuel assemblies in the offsite storage facility.

4.2 Spent Fuel Pool Cleanup System

The spent fuel pool cleanup system is part of the pool cooling system. It consists of a bypass flow (400 gpm) that passes through a 3 micron cartridge type filter followed by a mixed bed ion exchange demineralizer followed by a second 3 micron filter. There is also a separate skimmer system to remove surface dust and debris from the SFP. This cleanup system is similar to such systems at other nuclear plants which maintain concentrations of radioactivity in the pool water at low levels.

We expect only a small increase in radioactivity released to the pool water as a result of the proposed modification. We therefore conclude the spent fuel pool cleanup system is adequate for the proposed modification and will keep the concentrations of radioactivity in the pool water to acceptably low levels.

4.3 Radioactive Material Released to the Atmosphere

With respect to releases of gaseous materials to the atmosphere, the only radioactive gas of significance which could be attributable to storing additional fuel assemblies for a long period of time would be the noble gas radionuclide Krypton-85 (Kr-85). Experience has demonstrated that after spent fuel has decayed 4 to 6 months, there is no longer a significant release of fission products, including Kr-85, from stored fuel containing cladding defects.

For the NA-1&2 spent fuel pool modification, an average of 68 fuel assemblies for each NAPS Unit are expected to be stored following each refueling. In addition, approximately 500 fuel assemblies from the Surry Nuclear Generating Station may also be stored in the NA-1&2 spent fuel pool. Since space must be reserved to accommodate a complete reactor core unloading operation (157 fuel assemblies), the useful pool capacity is 1580 fuel assemblies. Allowing for the stored Surry fuel and for NA-1&2, full core storage capability will be maintained until after the sixteenth refueling cycle estimated for 1992. Up to this date, the oldest spent fuel will have been stored for approximately 13 years.

We assumed that all of the Kr-85 that is going to leak from defected fuel is going to do so in the interval between refuelings. The assumption is conservative and maximizes the amount of Kr-85 to be released. Our calculations show that the maximum expected release of Kr-85 from one refueling cycle (68 assemblies) is approximately 124.3 curies (see Table 4-1). Spent fuel discharges from both units are expected to yield an annual average release of 166 curies/year for Kr-85. This is not significant when compared to the projected 5700 curies per year of noble gas releases for the combined units from all other sources (FES Addendum 1 dated November 1976). Accordingly, the enlarged capacity of the pool has no significant effect on the greatest release rate of Kr-85 to the atmosphere each year. Thus, we conclude that the proposed modifications will have insignificant effects on offsite exposures.

Iodine-131 releases from spent fuel assemblies to the spent fuel pool water will not be significantly increased because of the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings for each unit.

Most of the tritium in the spent fuel pool water results from activation of boron and lithium in the primary coolant and this will not be affected by the proposed changes. A relatively small amount of tritium is contributed during reactor operation by fissioning of reactor fuel and subsequent diffusion of tritium through the fuel and the Zircaloy cladding. Tritium release from the fuel essentially all occurs while the fuel is hot, that is, during operations and, to a limited extent, shortly after shutdown. Thus, expanding spent fuel pool capacity will not increase the tritium activity in the spent fuel pool.

Storing additional spent fuel assemblies is not expected to increase the bulk water temperature during normal refuelings above the 150°F used in the design analysis. Therefore, it is not expected that there will be any significant change in the annual release of tritium or iodine as a result of the proposed modifications from that previously evaluated in the FES.

4.4 Solid Radioactive Wastes

The concentration of radionuclides in the pool water is controlled by the filters and the demineralizer and by decay of short-lived isotopes. The activity is highest during refueling operations when reactor coolant water is introduced into the pool and decreases as the pool water is processed through the filters and demineralizer. The increase of radioactivity, if any, due to the proposed modification should be minor because of the capability of the cleanup system to continuously remove radioactivity in the spent fuel pool water to acceptable levels (See Section 4.2).

The licensee states that the amount of solid waste presently being generated by the spent fuel pool cleanup system (i.e., approximately 22 cubic feet every year) is approximately 10 percent of the station total solid radioactive waste. The licensee does not expect that this spent fuel pool modification will result in any significant increase in this amount of solid waste generated from the spent fuel pool cleanup system. We agree with the licensee and note that, should there be an increase in spent fuel pool resin waste generation, the total would still be within those values estimated in the FES.

In addition to the small increase in the resin generated waste, it is estimated that activities associated with the loading/unloading of 500 Surry spent fuel assemblies will generate approximately 15,000 cubic feet of compressible solid waste such as rags, clothing, mop-heads, etc. This waste is estimated to contain a maximum of 27 curies for the loading/unloading of 500 Surry assemblies. For comparison, the 1981 annual shipment of solid waste from NA-1&2 was 10,700 cubic feet containing 2620 curies. This one time increase of solid waste (15,000 cubic feet) is 2.5 percent of the total solid waste estimated to be generated over the lifetime of NA-1&2 (approximately 428,000 cubic feet). Therefore, this one time increase in solid waste associated with loading/unloading activities of 500 Surry spent fuel assemblies should not burden waste disposal sites and will not have any significant impact on the environment.

The proposed modifications will require the removal of the presently inplace spent fuel racks. These spent fuel racks are contaminated and will be disposed of as low level waste. The exact disposal method had not yet been determined by the licensee. The licensee is considering two methods for disposal of the present spent fuel racks. One method would be the decontamination of fuel racks which would dispose only of those portions of the racks which could not be decontaminated. The other method would involve the cutting up (volume reduction) of the fuel racks for off-site disposal. The licensee estimates that the decontamination method would generate 2000 cubic feet of low level radioactive waste. The volume reduction method would generate 10,000 cubic feet of low level waste.

Based on the 1981 yearly total of solid waste (10,700 cubic feet) averaged over the lifetime of NA-1&2, the decontamination or volume reduction method would increase the total waste volume shipped off-site by one-half of one percent or 2.3 percent, respectively. Thus, the use of either method will not result in any significant additional impact to the environment.

The licensee estimates that the total curie content generated from either method would be 4 to 5 curies. This estimated curie content for rack disposal is approximately two-tenths of one percent of the total radioactive waste (2,620 curies) generated at NA-1&2 for the year 1981. Therefore, should the present racks be ultimately shipped to a burial site, this additional quantity of solid waste should not have any significant impact on the environment. Nevertheless, the licensee is requested to submit its finalized plans for rack disposal to the NRC for final approval.

4.5 Radioactive Material Released to Receiving Waters

Since the spent fuel pool cooling and cleanup system operates as a closed system, only water originating from cleanup of spent fuel pool floors and resin sluice water need be considered as potential sources of radioactivity.

It is expected that neither the quantity nor activity of the floor cleanup water will change as a result of this modification. The spent fuel pool demineralizer resin removes soluble radioactive material from the spent fuel pool water. These resins are periodically sluiced with water to the spent resin storage tank. The amount of radioactivity on the spent fuel pool demineralizer resin may increase slightly due to the additional spent fuel in the pool, but the soluble radioactive material would be retained on the resins. If any radioactive material is transferred from the spent resin to the sluice water, it will be removed by processing through the liquid radwaste system. Therefore, because the liquid waste processing system captures radioactive material, it is not expected that any additional radioactivity will be released to the environment resulting from the proposed modification.

4.6 Radiological Impact to Plant Workers

The proposed increase in storage capacity of the spent fuel pool would not affect significantly the radiological impact to the work force. The average dose to plant workers at NA-1&2, over the years 1979 through 1982, has averaged about 816 person-rems for both units/year.* The total projected worker dose for the proposed modifications is about 14 person-rems (including disposal of the present storage racks), which is about 1.7 percent of the normal annual rate.

In addition, the proposed loading/unloading of spent fuel shipping casks will increase the annual worker dose by an estimated 16.8 person-rem (based on 500 shipments over a 5 year period). This increase in annual worker dose represents only a small fraction (about 2 percent) of the normal annual rate.

Based on the above proposed actions, we find the total projected worker dose to be about 31 person-rems which is about 3.7 percent of the normal annual rate. We find this 3.7 percent increase in annual dose to plant workers to not be significant.

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*The average dose is the average total dose for both units, and is taken from "Occupational Radiation Exposure at Commercial Nuclear Power Reactors 1982", NUREG-0713, Vol. 4, December 1983.

4.7 Radiological Impacts to the Population

The proposed increase of the storage capacity of the spent fuel pool would not create a significant additional radiological impact to the population. The additional total body dose that might be received by an individual at the site boundary and by the population within a 50 mile radius is estimated to be less than 0.1 mrem/yr and less than 0.1 person-rem/yr, respectively. These doses are extremely small compared to the fluctuations in the annual dose this population receives from background radiation. The population dose represents an increase of less than 1 percent of the dose previously evaluated in the FES for NA-1&2. We find the dose to the population resulting from the proposed action to be not significant.

Table 4-1

Spent Fuel Pool Modifications

Estimated Release Rate of Kr-85

North Anna, Unit Nos. 1 & 2

Core = 157 fuel assemblies

Single Refueling = 68 core assemblies per unit per 18 months

Cladding = Zircaloy-4

Burnup = approximately 36,000 MWd/MTu

Weight of UO_2 in Core = 82.2 MT of UO_2 or 72.4 MTu

Escape rate Coeff. of KR-85 = 6.5×10^{-8} sec

Fission Yield of Kr-85 = 0.0034

Present Capacity = 966 fuel assemblies, approx. 10 years

Future Capacity = 1737 fuel assemblies; 13-18 years (depends on the amount of Surry fuel stored at North Anna)

Failed Fuel Fraction (NUREG-0017) = .0012

Half-life (Kr-85) = 10.7 years

Amt Kr-85 in fuel $\frac{\text{Production rate}}{\text{decay + leakage}}$

$$\text{Production Rate} = \frac{\underset{\text{atoms/f}}{0.0034} \times \underset{\text{f/MWsec}}{3.12 \times 10^{16}} \times 2775 \text{ MWT}}{72.4 \text{ MTu}}$$

$= \underline{4.0 \times 10^{15}}$ atoms/MTu sec

(decay = 2.05×10^{-9}/sec, leak = 6.5×10^{-8}/sec)

Amt Kr-85 in fuel $\underline{5.96 \times 10^{22}}$ atoms/MTu

3302 Curies/MTu

This model assumes that all KR-85 in the failed fuel assemblies will be released before the spent fuel is removed from the pool.

Simple case: All Kr-85 escape between refueling =

$$3302 \text{ curie/MTu} \times \frac{72.4 \text{ MTu}}{157 \text{ oss}} \times \frac{68 \text{ oss}}{\text{refuel}} \times .0012 = 124.3 \text{ curies/refueling}$$

For the two units, the average spent fuel input yields

$$124.3 \text{ curies/refuel} \times \frac{2 \text{ refuelings}}{18 \text{ months}} \times \frac{12 \text{ months}}{\text{yr}} = 165.3 \text{ curies/year}$$

5.0 Environmental Impacts of Postulated Accidents

5.1 Cask Drop Accidents

We have reviewed the licensee's March 22, 1982 submittal addressing NUREG-0612 requirements that the fuel building trolley (1 MH CR 15) used for moving spent fuel casks does not move over stored fuel. The hook centerline is capable of movement to 5 feet 4 inches from the edge of the fuel pool. The west wall of the fuel pool separates the spent fuel cask storage area from the fuel pool. Only during the movement of spent fuel casks into and out of the fuel building are the casks raised above the top of the fuel pool wall. The centerline of the casks during this movement can be no closer than 1 foot 10 inches from the outside edge of the pool wall. The trolley is equipped with eddy current brakes, dual load holding brakes and "dead man" controls. In addition, the lift height is limited to one foot. These characteristics greatly reduce the likelihood of occurrence of a cask drop, obviating the need for consideratior of the radiological consequences of an accident in which a dropped cask would impact stored fuel. Therefore, we conclude that an analysis of the radiological consequences of a cask drop accident is not required.

5.2 Spent Fuel Pool Gate Drop Accidents

We have reviewed the licensee's March 22, 1982 submittal addressing NUREG-0612 issues that the Fuel Building Movable Platform with Hoists (1-MH-FH-13), which is used to move the fuel cavity gates, is designed to be maneuvered over the spent fuel pool, the fuel transfer canals, and the new fuel handling and storage area as required during fuel handling operations. The movement of the platform is not restricted by electrical interlocks or mechanical stops. Technical Specifications prohibit the movement of loads in excess of 2,500 pounds from travel over irradiated fuel assemblies in the spent fuel pit and the licensee has proposed that plant procedures be revised to prohibit the handling of loads in excess of 2,000 pounds over spent fuel. In addition, administrative procedures will require that the top of the fuel cavity gates be secured to the top of the fuel pool wall by chains during movement of the gates to ensure that the gates, if dropped, will be prevented from tumbling into the fuel pool and damaging the spent fuel racks. Therefore, movement of the spent fuel pool gate should not result in an accident that could result in offsite radiological consequences.

5.3 Fuel Handling Accidents

The licensee in its August 1982 submittal states that the movement of the racks into position will either be done with a special temporary crane or by utilizing special rigging on the movable platform with hoist. The rig features remotely actuated positive capture devices, which preclude accidently dropping a rack during handling. All movement of spent fuel and spent fuel racks will be controlled by administrative procedures which will prohibit movement of the spent fuel racks over locations in the pool where fuel is stored. Therefore, the maximum loads which may be transported over spent fuel in the pool will be limited to that of a single assembly. The proposed spent fuel pool modification does not, therefore, increase radiological consequences of fuel handling accidents considered in the staff Safety Evaluation of June 1976, because this kind of accident would still result in, at most, release of the gap activity of one fuel assembly due to the limitations on available impact kinetic energy.

5.4 Conclusions

Based on the above, we conclude that the radiological consequences of accidents involving fuel handling accidents related to expansion of the spent fuel pool storage capacity at NA-1&2 meet the guidelines of 10 CFR Part 100, and are, therefore, acceptable.

6.0 Environmental Impact of the Transshipment of Spent Fuel From Surry to North Anna

The environmental impact of the transportation activity associated with the proposed transshipment of spent fuel from Surry to NAPS is within the scope of Table S-4 in 10 CFR 51.52 and therefore need not be addressed on a site-specific basis. The following Table compares pertinent parameters for the proposed transshipment from Surry to North Anna with the parameters used in WASH-1238 (Ref. 1) for calculating the environmental impacts contained in Table S-4.

Parameter	WASH-1238 (Ref. 1)	Proposed Transshipment (Surry to North Anna)
No. of shipments per year for two units	120	40 (Ref. 4)
Decay (cooling) time before shipment	150 days	730 days (Ref. 2)
Distance shipped (one way)	1,000 miles	177 miles (maximum) 159 miles (preferred route) (Ref. 3)

Parameter	WASH-1238 (Ref. 1)	Proposed Transshipment (Surry to North Anna)
Shipment duration	3 days	4 hr. 20 min (maximum) (Ref. 3)
Stops	Refueling; rest	None required

Comparing the proposed transshipment with the parameters used in WASH-1238, the radiological impact would be less by (1) a factor of 3 for number of shipments, (2) a factor of about 2.5 for decay time (see letter dated November 7, 1983, from C. V. Parks, Union Carbide Corporation, Nuclear Division, to R. H. Odegaarden, NRC, Reference 5), and (3) a factor of about 6 for distance shipped which gives a total reduction by a factor of about 45. It is noted that no credit is taken for shorter shipment duration or fewer stops. From these comparisons, the staff concludes that the radiological impact on the environment would be less by a factor of at least 30 than that shown in Table S-4 and accordingly, the impact would be well within the scope of Table S-4.

7.0 Summary

The Final Generic Environmental Impact Statement (FGEIS) on Handling and Storage of Spent Light Water Power Reactor Fuel concluded that the environmental impact of interim storage of spent fuel was negligible and the cost of the various alternatives reflects the advantage of continued generation of nuclear power with the accompanying spent fuel storage. Because of the differences in SFP designs, the FGEIS recommended licensing SFP expansion on a case-by-case basis. For

NA-1&2, the expansion of the spent fuel storage capacity to accommodate both NA-1&2 and 500 Surry spent fuel assemblies will not create any significant additional radiological effects. The additional total body dose that might be received by an individual at the site boundary and the estimated dose to the total body of the population within a 50-mile radius of the plant is less than 0.1 mrem per year and 0.1 person-rem per year, respectively. These doses are extremely small compared to the fluctuations in the annual dose this population receives from background radiation. This population dose represents an increase of less than 1 percent of the dose previously evaluated in the FES for NA-1&2. The occupational radiation dose to the work force engaged in the modification of the spent fuel storage racks (including present rack disposal) and the loading/unloading of 500 Surry spent fuel assemblies is estimated by the licensee to be 31 person-rem. This is a small fraction of the total person-rems from occupational dose at NA-1&2. The small increase in radiation dose should not affect the licensee's ability to maintain individual occupational dose within the limits of 10 CFR Part 20, and as low as reasonably achievable. Finally, pursuant to 10 CFR 51.52, the radiological impact to the environment related to the transshipment of 500 Surry spent fuel assemblies from Surry to NA-1&2 is well within the scope of Table S-4, and is therefore acceptable.

8.0 Basis and Conclusion for Not Preparing an Environmental Impact Statement

The staff has reviewed this proposed facility modification relative to the requirements set forth in 10 CFR Part 51 and the Council on Environmental Quality's Guidelines, 40 CFR 1500.6. Based on this assessment, we propose to find that the actions specified will not either separately or combined significantly impact on the quality of the human environment. These actions are:

Item 1 The installation of neutron absorber spent fuel storage racks in the North Anna Units No. 1 and No. 2 spent fuel pool which would increase the spent fuel storage capacity from the present 966 assemblies to 1737 assemblies.

Item 2 The storage of up to 500 spent fuel assemblies from the Surry Power Station Units No. 1 and No. 2 in the spent fuel pool at the North Anna Power Station Units No. 1 and No. 2.

Item 3 The transshipment of 500 Surry spent fuel assemblies from the Surry Power Station Units No. 1 and No. 2 to the North Anna Power Station, Units No. 1 and No. 2.

The staff has concluded that these actions involve no significant change in types or significant increase in the amounts of any effluents that may be released offsite and there is no significant increase in individual or cumulative occupational radiation exposure. Therefore, the staff has determined, pursuant to 10 CFR 51.31, that an environmental impact statement need not be prepared for Items 1, 2 and 3 specified above.

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APPENDIX A

Chronology of Environmental Assessment Review

Regarding

Spent Fuel Pool Expansion and Storage

of Surry Power Station Spent Fuel At

The North Anna Power Station

NOTE: Documents referenced in this chronology are available for public inspection and copying for a fee at the NRC Public Document Room 1717 H Street, N.W., Washington, D.C.; and at the Public Document Rooms located at Board of Supervisors Office, Louisa County Courthouse, Louisa, Virginia 23093 and the Alderman Library, Manuscripts Department, University of Virginia, Charlottesville, Virginia 22901.

July 13, 1982	Letter from R. H. Leasburg (licensee) to H. R. Denton, NRC, for a license amendment to permit the storage of 500 Surry spent fuel assemblies at NAPS.
July 13, 1982	Letter from R. H. Leasburg (licensee) to Robert F. Bernett (sic).
July 28, 1982	Letter from Theodore S. Sherr, NRC, to R. H. Leasburg (licensee).
August 20, 1982	Letter from R. H. Leasburg (licensee) to H. R. Denton, NRC, for a license amendment to modify spent fuel storage to 1737 fuel assemblies.

October 21, 1982	Letter from R. H. Leasburg (licensee) to H. R. Denton, NRC, transmitting additional information on spent fuel pool heat loads.
June 16, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, transmitting responses to NRC questions.
July 19, 1983	Letter from R. Clark, NRC, to W. L. Stewart (licensee) requesting information regarding the transshipment of fuel from Surry to North Anna.
July 25, 1983	Letter from R. Clark, NRC, to W. L. Stewart (licensee) requesting additional information regarding spent fuel storage expansion.
September 13, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, transmitting responses to NRC request for additional information.
November 10, 1983	Letter from J. R. Miller, NRC, to W. L. Stewart requesting additional information regarding spent fuel storage capacity.
November 23, 1983	Letter from W. L. Stewart to H. R. Denton, NRC, advising when additional information will be provided by licensee.
December 6, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing additional information regarding spent fuel capacity expansion.
December 6, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing additional information regarding increase in fuel capacity.

December 14, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, correcting typographical errors in the licensee letter dated December 6, 1983.
December 14, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing clarification for neutron absorber spent fuel rack.
December 29, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing additional information regarding spent fuel capacity expansion.
December 29, 1983	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing additional information of proprietary nature regarding neutron absorber racks.
April 10, 1984	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing additional information regarding spent fuel expansion.
May 8, 1984	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing present status of NA-1&2 and Surry-1&2 spent fuel storage capabilities.
May 18, 1984	Letter from W. L. Stewart (licensee) to H. R. Denton, NRC, providing additional information regarding low level radioactive waste.

Appendix B

References

(1) USAEC Report WASH-1238, "Environmental Survey of Transportation of Radioactive Materials To and From Nuclear Power Plants", December 1972.

(2) Letter from R. H. Leasburg, VEPCO, to Harold R. Denton, USNRC, July 13, 1982.

(3) Letter from R. H. Leasburg, VEPCO, to Robert F. Burnett, USNRC, July 13, 1982.

(4) Letter from W. L. Stewart, VEPCO, to Harold R. Denton, USNRC, October 28, 1983.

(5) Letter from C. V. Parks, Union Carbide Corporation, Nuclear Division, to R. H. Odegaarden, USNRC, November 7, 1983.