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

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

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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 to 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 fro 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 to 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

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

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

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

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

Amt Kr-85 in fuel 5.96×10^{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

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

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 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.

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.