State Archaeological Research Center, Rapid City, SD.

A detailed assessment of the human remains was made by South Dakota State Archaeological Research Center and Office of the State Archeologist of Iowa professional staff in consultation with representatives of the Three Affiliated Tribes of the Fort Berthold Reservation.

In 1934, human remains representing one individual likely to have been recovered from the Evart's Village site (39WW204), Walworth County, SD during Works Project Administration road construction. No known individual was identified. The two associated funerary objects consist of a white glass pony bead and a rifle bullet, identified as possibly a .44-60 calibre Peabody, Remington, or Sharps.

In 1990, these human remains were found in the collections of the Conger House Museum in Washington, IA and transferred to the Office of the State Archeologist of Iowa. Museum documentation suggests these remains were recovered from the Evart's Village site on Fred Brazel's land near Evart, SD and given to his brother-in-law, Thomas Royster of Muscatine, IA. Mr. Royster may have donated these remains to the Conger House Museum, as Washington, IA is near Muscatine. In 1952, an interview with Mrs. Fred Brazel revealed that these human remains were possibly interred as a primary flexed or secondary bundle burial, placed face up on top of a layer of cut willow twigs.

Based on skeletal morphology and associated funerary objects, these individuals have been determined to be Native American. Based on the associated funerary objects, manner of interment, and geographical location, the Evart's Village site has been identified as a post–1770 Arikara or Mandan village. Consultation with representatives of the Three Affiliated Tribes indicates there were Arikara and Mandan villages in this area of South Dakota during the post contact period.

Based on the above mentioned information, officials of the South Dakota State Archaeological Research Center have determined that, pursuant to 43 CFR 10.2 (d)(1), the human remains listed above represent the physical remains of one individual of Native American ancestry. Officials of the South Dakota State Archaeological Research Center have also determined that, pursuant to 43 CFR 10.2 (d)(2), the two objects listed above are reasonably believed to have been placed with or near individual human remains at the time of death or later as part of the death rite or ceremony. Lastly, officials of the South Dakota State Archaeological Research Center have determined that, pursuant to 43 CFR 10.2 (e), there is a relationship of shared group identity which can be reasonably traced between these Native American human remains and associated funerary objects and the Three Affiliated Tribes of the Fort Berthold Reservation.

This notice has been sent to officials of the Three Affiliated Tribes of the Fort Berthold Reservation. Representatives of any other Indian tribe that believes itself to be culturally affiliated with these human remains and associated funerary objects should contact Renee Boen, Curator, South Dakota State Archaeological Reserch Center, P.O. Box 1257, Rapid City, SD 57709-1257; telephone: (605) 394–1936, before August 22, 1999. Repatriation of the human remains and associated funerary objects to the Three Affiliated Tribes of the Fort Berthold Reservation may begin after that date if no additional claimants come forward.

Dated: July 6, 1998.

Francis P. McManamon,

Departmental Consulting Archeologist, Manager, Archeology and Ethnography Program.

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NUCLEAR REGULATORY COMMISSION

[Docket Nos. 50–280 and 50–281]

In the Matter of Virginia Electric and Power Company; Surry Power Station, Units 1 and 2; Exemption

I

The Virginia Electric and Power Company (VEPCO, the licensee) is the holder of Facility Operating License No. DPR–32 and Facility Operating License No. DPR–37, which authorize operation of the Surry Power Station, Units 1 and 2. The licenses provide that the licensee is subject to all rules, regulations, and orders of the U.S. Nuclear Regulatory Commission (NRC or the Commission) now or hereafter in effect.

The facility consists of two pressurized-water reactors at the licensee's site located in Surry County, Virginia.

II

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 70.24, ''Criticality Accident Requirements,'' requires that each licensee authorized to possess special nuclear material shall maintain a criticality accident monitoring system in each area in which such material is handled, used, or stored. Sections 70.24 (a)(1) and (a)(2) specify detection and sensitivity requirements that these monitors must meet. Section 70.24(a)(1) also specifies that all areas subject to criticality accident monitoring must be covered by two detectors. Section 70.24(a)(3) requires licensees to maintain emergency procedures for each area in which this licensed special nuclear material is handled, used, or stored, and provides (1) that the procedures ensure that all personnel withdraw to an area of safety upon the sounding of a criticality accident monitor alarm, (2) that the procedures must include drills to familiarize personnel with the evacuation plan, and (3) that the procedures designate responsible individuals for determining the cause of the alarm and placement of radiation survey instruments in accessible locations for use in such an emergency. Section 70.24(b)(1) requires licensees to have a means by which to quickly identify personnel who have received a dose of 10 rads or more. Section 70.24(b)(2) requires licensees to maintain personnel decontamination facilities, to maintain arrangements for a physician and other medical personnel qualified to handle radiation emergencies, and to maintain arrangements for the transportation of contaminated individuals to treatment facilities outside the site boundary. Section 70.24(c) exempts Part 50 licensees from the requirements of 10 CFR 70.24(c) for special nuclear material used or to be used in the reactor. Subsection 70.24(d) states that any licensee who believes that there is good cause why he should be granted an exemption from all or part of 10 CFR 70.24 may apply to the Commission for such an exemption and shall specify the reasons for the relief requested.

III

On August 21, 1997, the NRC granted an exemption from the requirements of 10 CFR 70.24 reflecting the licensee's use of fuel enriched to 4.1 weight percent U235. By letter dated January 14, 1998, VEPCO requested a revised exemption from 10 CFR 70.24(a) based on the use of fuel enriched to 4.3 weight percent U235. The Commission has reviewed the licensee's submittal and has determined that inadvertent criticality is not likely to occur in special nuclear materials handling or storage areas at Surry Power Station, Units 1 and 2. The quantity of special nuclear material other than fuel that is stored on site is small enough to preclude achieving a critical mass.

The purpose of the criticality monitors required by 10 CFR 70.24 is to ensure that if a criticality were to occur during the handling of special nuclear material, personnel would be alerted to that fact and would take appropriate action. Although the staff has determined that such an accident is not likely to occur, the licensee has radiation monitors, as required by General Design Criteria 63, in fuel storage and handling areas. These monitors will alert personnel to excessive radiation levels and allow them to initiate appropriate safety actions. The low probability of an inadvertent criticality together with the licensee's adherence to General Design Criterion 63 constitute good cause for granting an exemption to the requirements of 10 CFR 70.24(a).

IV

The Commission has determined that, pursuant to 10 CFR 70.14, this exemption as revised is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest; therefore, the Commission hereby grants the following revised exemption:

The Virginia Electric and Power Company is exempt from the requirements of 10 CFR 70.24(a) for the Surry Nuclear Power Station, Unit 1 and Unit 2.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this revised exemption will have no significant impact on the quality of the human environment (63 FR 38196).

This revised exemption is effective upon issuance.

Dated at Rockville, Maryland this 15th day of July 1998.

For the Nuclear Regulatory Commission.

Samuel J. Collins,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 98–19539 Filed 7–21–98; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

[Docket No. 50–244]

Rochester Gas and Electric Corporation; R.E. Ginna Nuclear Power Plant; Environment Assessment and Finding of No Significant Impact

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. DRP–18, issued to Rochester Gas and Electric Corporation (the licensee), for operation of the R.E. Ginna Nuclear Power Plant, located in Wayne County, New, York.

Identification of the Proposed Action

The proposed action would modify the spent fuel pool (SFP) by replacing the three Region 1 rack modules with seven new borated stainless steel rack modules scheduled for implementation in 1998. Six new peripheral modules would be added at some future date. Two of the seven new modules planned to be installed in 1998 would be designated as part of Region 2, effectively increasing the Region 2 area. The other five new modules would compose Region 1, resulting in a total of 294 storage positions in Region 1. Region 2, with 1075 storage positions, would consist of three rack types, Type 1, Type 2, and Type 4. Type 1 cells are the Boraflex cells that form Region 2 for the existing license. Two racks of Type 2 cells, containing borated stainless steel (BSS) absorber plates, would be added to increase the storage capacity of Region 2. In addition, the capacity of Region 2 could be increased in the future by the addition of Type 4 racks, which also contain BSS absorber plates. The amendment would also increase the boron concentration from 300 ppm to 2300 ppm.

The proposed action is in accordance with the licensee's application for amendment dated March 31, 1997, as supplemented June 18, 1997, October 10, 1997, November 11, 1997, December 22, 1997, January 15, 1998, January 27, 1998, March 20, 1998, April 23, 1998, April 27, 1998, and May 8, 1998.

The Need for the Proposed Action

The proposed action would modify the spent fuel pool to accommodate storage of spent fuel until the expiration of the Ginna Station license in 2009. The current configuration of the Ginna spent fuel storage pool consists of two regions. Region 1 consists of stainless steel racks with 176 storage locations in a checker board pattern. Region 2 consists of stainless steel racks with boraflex and with 840 storage locations. This provides a total of 1016 storage locations. The proposed amendment would replace the Region 1 racks with borated stainless steel racks. Two locations are proposed in Region 1, one with borated stainless steel that would accommodate 187 storage locations and one with borated stainless steel in a checker board pattern that would accommodate 292 storage locations. This would provide a total of 1319 storage locations which would provide enough storage locations for storage of spent fuel beyond the expiration of the license in 2009.

Environmental Impacts of the Proposed Action

Radioactive Waste Treatment

The Ginna Nuclear Power Plant uses waste treatment systems designed to collect and process gaseous, liquid, and solid waste that might contain radioactive material. These radioactive waste treatment systems are evaluated in the Final Environmental Statement (FES) dated December 1973. The proposed rerack will not involve any change in the waste treatment systems described in the FES.

Gaseous Radioactive Wastes

The only radioactive gas of significance that could be attributable to storing additional spent fuel assemblies for a longer 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 spent fuel containing cladding defects. The licensee has stated that the Kr-85 noble gases are not normally released from the Auxiliary Building on a continuous basis and enlarging the storage capacity of the SFP will have no effect on the average annual quantities of Kr-85 released to the atmosphere.

Iodine-131 released from spent fuel assemblies to the SFP water will not be significantly increased due to the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings.

The amount of tritium in the SFP water will not be affected by the proposed changes. Most of the tritium in the SFP water results from activation of boron and lithium in the primary coolant. A relatively small amount of tritium is produced during reactor operation by the fission process within the reactor fuel. The subsequent diffusion of the tritium through the fuel and cladding represents a small contribution to the total amount of tritium in the SFP water. Tritium releases from the fuel assemblies occur mainly during reactor operation and, to a limited extent, shortly after shutdown. Thus, expanding the SFP capacity will not increase the tritium activity in the SFP.

Most airborne releases of tritium and iodine from nuclear power plants result during refuelings from evaporation of reactor coolant, which contains tritium and iodine in higher concentrations than in the SFP. The storage of additional spent fuel assemblies in the SFP is not expected to increase the SFP