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SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION UNITS 1 AND 2

SURRY COUNTY, VIRGINIA

DOCKET NOS. 50-280 AND 50-281

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1.0 INTRODUCTION

The Virginia Electric and Power Company (hereinafter referred to as Vepco or the applicant) by application dated March 20, 1967, and as subsequently amended, requested a license to construct and operate two pressurized water reactors, identified as Surry Power Station Units 1 and 2 at a site in Surry County, Virginia, on the south bank of the James River, 4.7 miles northwest of the corporate boundary of Newport News, Virginia. The Atomic Energy Commission reported the results of its review prior to construction in a Safety Evaluation dated May 2, 1968. Following a public hearing before an Atomic Safety and Licensing Board in Williamsburg, Virginia on May 28, 1968, the Director of Reactor Licensing issued provisional construction permit numbers CPPR-43 and CPPR-44 for Units 1 and 2, respectively, on June 25, 1968.

On January 21, 1970, the applicant filed, as Amendment No. 12, the Final Safety Analysis Report required by Section 50.34(b) of Chapter 10 of the Code of Federal Regulations (10 CFR) as a prerequisite to obtaining an operating license for each unit.

The operating license applications are for a power level of 2441 megawatts thermal (MWt), the same thermal power requested in the construction permit applications. Except for the performance of the emergency core cooling system, our evaluation of the engineered safety features and the accident analyses have been performed at the ultimate anticipated power level of 2546 MWt. Before operation at any power level above 2441 MWt is authorized, the regulatory staff will perform a safety evaluation to assure that the core can be operated safely at the higher power level.

Our technical safety review with respect to issuing operating licenses for the Surry Station has been based on the applicant's Final Safety Analysis Report (amendment 12) and subsequent amendments 13 through 32 inclusive, all of which are available for review at the Atomic Energy Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C. and at the Swim Library, College of William and Mary, Williamsburg, Virginia. In the course of our review of the material submitted, we held a number of meetings with representatives of the applicant, the nuclear steam system supplier, Westinghouse Electric Corporation, and the

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applicant's architect-engineer, Stone & Webster Engineering Corporation, to discuss the plant design, construction, and the proposed operation. A chronology of our review is attached as Appendix A to this evaluation.

This safety evaluation report summarizes the results of the technical evaluation of the Surry Power Station Units 1 and 2 performed by the Commission's regulatory staff and delineates the scope of the extensive technical details considered in evaluating the radiological safety aspects of the proposed operation of the facility. In accordance with the provisions of 10 CFR Part 50, Appendix D, of the Commission's Regulations, a subsequent report will set forth the considerations of the environmental impact of the proposed operation of the facility.

Our evaluation included a technical review of the information submitted by the applicant particularly with regard to the following principal matters:

1. We evaluated the population density and land use characteristics of the site environs and the physical characteristics of the site, including seismology,

meteorology, geology, and hydrology to establish that these characteristics had been determined adequately and had been given appropriate consideration in the final design of the plant, and that the site characteristics are in accord with the Commission's siting criteria (10 CFR Part 100), taking into consideration the design of the facility including the engineered safety features provided.

2. We evaluated the design, fabrication, construction, and testing and performance characteristics of the plant structures, systems, and components important to safety to determine that they are in accord with the Commission's General Design Criteria, Quality Assurance Criteria, Safety Guides and other appropriate rules, codes, and standards, and that any departures from these criteria, codes, and standards have been identified and justified.

3. We evaluated the expected response of the facility to various anticipated operating transients and to a broad spectrum of accidents and determined that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed

those of all other accidents considered. We performed conservative analyses of these design basis accidents to determine that the calculated potential off-site doses that might result in the very unlikely event of their occurence would not exceed the Commission's guidelines for site acceptability given in 10 CFR Part 100.

4. We evaluated the applicant's engineering and construction organizations, plans for the conduct of plant operation, including the proposed organization, staffing and training program, the plans for industrial security, and the scope of planning for emergency actions to be taken in the unlikely event of an accident that might affect the general public, to determine that the applicant is technically qualified to safely operate the plant.

5. We evaluated the design of the systems provided for control of the radioactive effluents from the plant to determine that these systems are capable of controlling the release of radioactive wastes from the

facility within the limits of the Commission's regulations (10 CFR Part 20) and that the equipment provided is capable of being operated by the applicant in such a manner as to reduce radioactive releases to levels that are as low as practicable.

6. We evaluated the financial position of the applicant to determine that the applicant's financial resources are adequate to operate Surry Power Station Units 1 and 2 in accordance with the activities that would be permitted by the operating license.

Many features of the design of this plant are similar to those we have evaluated and approved previously for other reactors now under construction or in operation. To the extent feasible and appropriate, we have made use of our previous evaluations to expedite our review of those features that were shown to be substantially the same as those previously considered. Where this has been done, the appropriate sections of this report identify the other facilities involved. Our Safety Evaluation reports for these other facilities have been published and are available for public inspection at the Atomic Energy Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C.

In addition to our review, the Advisory Committee on Reactor Safeguards (ACRS) reviewed the application and met with both the applicant and the regulatory staff to discuss the facility. The ACRS Report to the Commission on the Surry Power Station Units 1 and 2, dated December 17, 1971, is attached as Appendix B to this Safety Evaluation.

Based on our evaluation of the application to construct and operate the facility, we have concluded that each unit of the Surry Power Station can be operated as proposed at power levels up to 2441 MWt without undue risk to the health and safety of the public. Subsequent to issuance of an operating license, the units will be required to operate in accordance with the terms of the operating license and the Commission's regulations under the surveillance of the Commission's regulatory staff.

2.0 FACILITY DESCRIPTION

Surry Power Station Units 1 and 2 each employs a nuclear steam supply system consisting of a pressurized water reactor and a 3-loop reactor coolant system. The reactor core is composed of fuel rods made of uranium dioxide pellets enclosed in

Zircaloy tubes with welded end plugs. The fuel rods are grouped and supported in assemblies. Initially, the reactor core will be loaded in three regions each utilizing fuel of a different enrichment of U-235. Water will serve as both the moderator and the coolant and will be circulated through the reactor vessel and the core by three coolant pumps. Isolation valves are provided in each of the three reactor coolant loops to permit isolation of a loop during operation. The reactor coolant water, heated by the reactor, will flow through the three steam generators (one in each loop) where heat will be transferred to the secondary (steam) system. The water will then flow back to the pumps to repeat the cycle. An electrically heated pressurizer will establish and maintain the reactor coolant pressure and provide a surge chamber and a water reserve to accommodate reactor coolant volume changes during operation.

Engineered safety features comprised of systems of cooling water sources, pumps, valves, and associated piping are provided to cool the reactor core should a loss-of-coolant accident occur.

The two nuclear steam supply systems will be housed in separate reinforced concrete containment structures. Each

containment structure will be maintained at subatmospheric conditions during operation by the periodic use of vacuum pumps. In the event of a loss-of-coolant accident (discussed in Section 3.1.9.2 below), the containment pressure would initially increase and subsequently decrease in a manner similar to that of other non-pressure suppression type of containment systems. However, since the containment normally operates at a pressure of approximately 10 psia, the pressure inside containment will decrease to below atmospheric conditions in a relatively short time following an accident. This will result in the cessation of leakage of radioactive materials from the containment.

An auxiliary building, located between the containment structures for Units 1 and 2, houses the radioactive waste treatment facilities, components of the engineered safety features, and various related auxiliary systems for each reactor unit. Both units share a common fuel handling facility which contains a spent fuel pool and a new fuel storage facility. Engineered safety feature equipment required to function for long periods following an accident is located immediately adjacent to the exterior wall of the containment wall on an extension of the containment base mat.

The steam and power conversion system is designed to remove heat energy from the reactor coolant in the three steam generators and convert it to electrical energy. The heat removed by the condensers will be discharged through the circulating water system to the James River.

The reactor will be controlled by control rod movement and regulation of the boron concentration in the reactor coolant. The control elements will be moved vertically within the core by individual control rod drive mechanisms. A reactor protection system that automatically initiates appropriate action whenever a plant condition monitored by the system approaches pre-established limits is provided. This reactor protection system will act to shut down the reactor, close isolation valves, and initiate operation of the engineered safety features should any or all of these actions be required.

The plant will be capable of being supplied with electrical power from two independent offsite power sources and will be provided with independent and redundant onsite emergency power supplies capable of supplying power to the engineered safety features.

3.0 IMPORTANT SAFETY CONSIDERATIONS

3.1 Suitability of the Site

3.1.1 General

In evaluating this reactor site in accordance with the Commission's siting guidelines, 10 CFR Part 100, we have considered the following aspects: the population density and land use characteristics of the site environs, the physical characteristics of the site, including seismology, meteorology, geology, and hydrology. The design of the facility, particularly the engineered safety features and the radioactive waste management systems, can be influenced by site characteristics and therefore, were considered as part of the site evaluation.

3.1.2 Site Description and Population Considerations

The Surry site consists of 840 acres located 7 miles due south of Williamsburg in Surry County, Virginia, on the Gravel Neck Peninsula which extends from the south bank of the James River. The nearest population center with a population of 25,000 or more is Newport News, Virginia, with the nearest corporate boundary of Newport News 4.7 miles southeast of the site. We consider this to be the population center distance

even though significant population density does not occur within 15 miles of the site. The nearest boundary of the exclusion area is located 1,650 feet (0.3 mile) from the containment buildings, and the nearest occupied residence is located 0.6 mile southwest of the containment buildings. The low population zone is defined by the applicant to be three miles. In 1966, the population of the low population zone was 122. The 1970 population of Newport News, the nearest population center, was 138,177.

On the basis of our evaluation of the population data, and of the calculated potential offsite doses that might result from a postulated design basis accident (presented in Section 3.1.9), we have concluded that the exclusion area radius, the low population zone distance, and the distance to the nearest population center meet the guidelines given in 10 CFR Part 100 and are acceptable.

3.1.3 Meteorology

The diffusion climatology of the Surry site is principally controlled by the marine influences of the Atlantic Ocean and Chesapeake Bay. To some degree, the local microclimatology will be affected by the James River into which the Gravel Neck Peninsula extends.

The applicant has collected meteorological data from two locations, one onsite (referred to as Surry), and the other located on Hog Island, north of the site at the tip of the Gravel Neck Peninsula. The Surry data cover an elapsed time of two years with 69% data recovery. Measurements of wind speed and wind direction were made 150 feet above the ground. Stability conditions were determined by the variability of wind direction. The Hog Island data cover an elapsed time of one year with 77% data recovery. Measurements were made 20 feet above the surface. Parameters measured were the same as for the Surry location.

We have evaluated these data and have concluded that the most appropriate basis for determining the short-term meteorology for an assumed ground level release is to use the Surry tower data and estimate stability conditions from the variability of wind direction, extrapolating the wind speeds at the 150 feet elevation of the tower to ground level. This analysis results in an estimate of Pasquill type F stability and a 1.5 meter per second wind speed for short-term releases. However, in view of the error band associated with estimating stability conditions by the variability of wind direction

method and considering the relatively poor data recovery at the Surry location, we consider that Pasquill Type F stability conditions with a 1.0 meter per second wind speed are more appropriate parameters to use as a conservative model for use in evaluating the radiological consequences of postulated accidents described in Section 3.1.9 of this report. The comments from our consultants at the National Oceanic and Atmospheric Administration (NOAA) which support our conclusion are attached as Appendix C to this Safety Evaluation.

3.1.4 Geology and Seismology

We and our consultants reviewed the geology and seismology of the site with respect to foundation conditions and intensity of earthquakes at the construction permit stage of our review. No new developments have occurred since that time to change our previous conclusion on the acceptability of these characteristics of the Surry Station site.

The reports of our consultants, the U. S. Coast and Geodetic Survey and the U. S. Geological Survey which were prepared during the construction permit stage of our review are attached as Appendices D and E respectively.

To provide the capability to determine the seismic response of the facility in the unlikely event of an earthquake, surveillance instrumentation for this purpose is provided. A seismic sensing and recording system, incorporating three remote triaxial accelerometers is installed at the Surry Power Station. The system will provide data on the frequency, amplitude, and phase relationship of the seismic response of the containment structures and provide data on the vibratory ground motion at the site. The data that would be obtained from this instrumentation would be employed in subsequently evaluating the effects of the seismic event on the continued safe operation the the facility. We have concluded that the Applicant's program for seismic instrumentation is in accord with AEC Safety Guide No. 12, Instrumentation For Earthquakes, and is acceptable.

3.1.5 Hydrology

The requirements for flood protection of the facility were based on flood studies conducted by the applicant and substantiated by an independent analysis. The probable maximum flood at the Surry site results from the probable

maximum hurricane. The applicant has calculated the probable maximum hurricane surge in the James River at the plant site utilizing the bathystrophic storm tide theory. The probable maximum hurricane would result in a storm surge of 22.3 feet above mean sea level (MSL). In addition, coincident wave runup on the east side of the peninsula will lead to a maximum water level of 28.2 feet above MSL on the embankment and 36.2 feet above MSL on the open water side of the pump house. Although this will result in overtopping the roof of the emergency pump house, the roof is capable of withstanding the hydrostatic loadings and the building is watertight.

The maximum still water level and wave runup on the west side of the peninsula following the probable maximum hurricane are calculated to be 23.0 feet above MSL and 1.3 feet, respectively. Critical equipment is protected to 26.5 feet above MSL. Thus, a margin of 2.2 feet is available between the maximum wave runup from the west and the protection level.

The applicant has calculated the maximum wave runup in the high level intake canal to be 9.5 feet. A minimum free board of 10 feet will be maintained.

We and our consultant, the Coastal Engineering Research Center, have independently reviewed the applicant's calculations of hurricane surge and wave runup and have concluded that the flood protection that will be provided for the facility is acceptable. The report of our consultant is attached as Appendix F.

3.1.6 Liquefaction Potential

In the event of earthquake-induced cyclic shearing of a soil mass, particles can become separated by films of liquid. If this occurs, all shearing strength is lost and the soil behaves as a heavy fluid, possibly resulting in the collapse or distortion of structures founded above it. In order for the particles to separate and liquify readily, soil density must be low, and grain size should be uniform. Because of the nature of the soil deposits at the Surry site, the potential for liquefaction was investigated. To reduce the potential for soil liquefaction, the applicant has installed sump pumps exterior to containment to reduce the piezometric levels under the Surry structures. We have evaluated the analysis performed by the applicant considering the measured depressed piezometric levels present. On this basis, we have concluded that the design has been carried out in a manner which makes the probability of liquefaction negligible. The potential for

liquefaction has also been evaluated by our consultant, Nathan M. Newmark Consulting Engineering Services. Its report, which supports our conclusion, is attached as Appendix G to this Safety Evaluation.

3.1.7 Environmental Radiological Monitoring

The applicant has presented two years of pre-operational environmental monitoring data, covering most of the principal pathways of exposure to man. The program includes (1) bimonthly sampling of the James River at five locations for gross beta activity and tritium; (2) semi-annual monitoring of four wells (two deep, two shallow) in the vicinity of the site for gross alpha and gross beta activity; (3) semi-annual monitoring of Chippokes Creek, the Williamsburg Reservoir, Newport News Reservoir, and the Smithfield Reservoir for soluble and insoluble gross alpha and gross beta activity; (4) monthly gross beta activity and semi-annual tritium monitoring of precipitation at Surry and at Newport News; (5) weekly gross beta activity and monthly gross alpha activity particulate monitoring at nine different locations; (6) monthly radiogas monitoring at ten different locations; (7) annual gamma scan and monitoring of corn, peanut, and soybean for gross beta activity and potassium-40, (8) semi-annual monitoring of coot for gross beta activity, potassium-40, strontium-90, and

iodine-131; (9) bimonthly gamma scan and monitoring of oysters from three separate locations for gross beta activity and potassium-40; (10) bimonthly gamma scan and monitoring of clams from four different locations for gross beta activity and potassium-40; (11) gamma scan and monitoring of crab for gross beta activity, and potassium-40 scan during the summer months; (12) bimonthly gamma scan and monitoring of silt from six different locations for gross beta activity and potassium-40; and (13) bimonthly monitoring of milk from five different locations for strontium-90, cesium-137, iodine-131, and calcium.

We have evaluated the data generated by the pre-operational environmental monitoring program and have concluded that it provides an adequate base upon which to determine the effect of radioactive releases from the plant. The applicant has stated that it is anticipated that this program will be continued during the operational phase of the station. The Technical Specifications require that the applicant conduct an environmental monitoring program similar in scope to that conducted during the pre-operational phase throughout the life of the plant.

Our consultant, the U. S. Department of the Interior, Fish and Wildlife Service, has reviewed the pre-operational and post-operational radiological monitoring program. It has recommended that the program be expanded to include representative species of carnivorous and herbivorous fishes and aquatic vegetation collected within 500 feet of the discharge point. The radiological monitoring program incorporated in the Technical Specifications includes these requirements. The report of the Fish and Wildlife Service is attached as Appendix H.

3.1.8 Radioactive Waste Management Systems

3.1.8.1 General

The radioactive waste systems of the Surry power station are designed to collect and process the liquid, gaseous, and solid wastes that are byproducts of station operation and that might contain radioactive materials. The systems are designed to accommodate the wastes produced during simultaneous operation of both Unit 1 and Unit 2 and are common to both units. The systems are sized on the assumption that both units are operating on a daily load-follow cycle between 100% and 50% of power using boric acid to adjust for reactivity changes in the core. Activity levels are based on the assumption that 1% of

the fuel rods in the core will have fuel cladding defects through which fission products may diffuse into the reactor coolant.

The applicant has indicated his intent to maintain and operate the waste treatment systems to keep radioactive releases as low as practicable. This intent is reflected in the Technical Specifications.

3.1.8.2 Liquid Waste System

The liquid radioactive waste treatment system consists of the tanks, piping, pumps, evaporators, process equipment, and instrumentation necessary to collect, process, store, analyze, monitor, and discharge all potentially radioactive wastes from the plant. Although the liquid wastes will be continuously collected, the process and storage systems will be sized so that processing of this waste will be done on a periodic basis. Thus, treated liquid wastes to be released will be handled on a batch-basis as required, permitting optimum control measures, and reducing the chance for an inadvertent discharge of radioactive liquid in excess of release limits. Prior to release of any treated liquid wastes, samples will be analyzed to determine the type and amount of radioactivity in a batch to assure conformance with release limits. Radiation

monitors will automatically terminate liquid waste discharges if the high radiation levels are detected in the discharge line.

Drain liquids originating from the reactor coolant system will be discharged to a primary drain transfer tank through a high pressure drain header. The liquids will then be pumped to the primary drain tank in the boron recovery system. The boron recovery system will degassify and store water so collected and further process it by evaporation, filtration, and demineralization to remove the boric acid from solution and to produce primary grade water. The applicant has estimated that 85% of the liquid processed by the boron recovery system will be recycled to the primary grade water system for reuse in the reactor coolant system. The remaining 15% will be further processed by the liquid waste treatment system prior to release to the James River. Other radioactive wastes generated in the plant will be sent directly to the liquid waste treatment system. All liquid wastes will be processed in accordance with their activity levels. High level liquid wastes will be processed by evaporation. The distillate will be sampled following evaporation and, if contaminated, will be further processed by flow through a mixed bed demineralizer and a filter or, if necessary, it will be re-evaporated. If

the activity is sufficiently low following sampling, discharge will be made to the discharge canal, and then to the James River. Low level liquid wastes will be filtered and then discharged through the discharge canal. As before, the discharge will be continuously monitored and if a high radiation level is detected, the valves in the discharge line will be automatically closed.

If the steam generator blowdown contains radioactivity, it will be processed through the liquid waste treatment system as necessary to keep releases as low as practicable.

The applicant has estimated that the annual quantities and concentrations of liquid radioactive wastes released to the circulating water discharge canal, based on continuous operation at the coolant design basis activity limit (1% failed fuel) in both reactor units for the entire year with no steam generator leakage will be approximately 2 curies per year gross activity excluding tritium, and 5500 curies of tritium per year. On an annual average basis, these releases would result in less than 1% of the 10 CFR Part 20 concentrations for all nuclides in the discharge canal (excluding tritium).

On the basis of our evaluation of the details of the liquid waste management system and our review of the applicant's analyses of the anticipated releases, we have concluded that the liquid waste management system provided for the Surry Power Station will be capable of limiting radioactive releases to a small fraction of the 10 CFR Part 20 limits. The Technical Specifications will require that the system be operated in such a manner that releases are as low as practicable.

3.1.8.3 Gaseous Waste Disposal System

The gaseous waste disposal system will process the fission product gases from contaminated reactor coolant fluids resulting from the operation of both units at the design basis reactor coolant activity limit. Gaseous wastes, primarily hydrogen, nitrogen, and minor amounts of fission product gases, will be removed from the reactor coolant letdown flow by the gas stripper in the boron recovery system. These gases will be processed in the gaseous waste disposal system, together with those collected by the various vents throughout the plant and from the liquid evaporator in the liquid waste disposal system. To reduce the volume of gases which must be stored, the gaseous waste disposal system will employ a catalytic hydrogen recombiner. Use of this recombiner will reduce the volume of gas to be handled by approximately a factor of 10.

Following recombination, waste gases will be sent to the two double-walled waste gas decay tanks. Each tank is designed to Class I (seismic) standards and is buried for tornado protection. The inner tank is fabricated from austenitic stainless steel in accordance with Section III of the ASME Boiler and Pressure Vessel Code, while the outer tank is fabricated from carbon steel in accordance with Section VIII of the same code. Overpressure relief protection is provided at the waste gas decay tanks in accordance with the requirements of Section III by bellows sealed pressure relief valves followed by rupture disc assemblies. Rupture disc bypasses with a higher pressure rating are provided around the relief valves. The use of the bellows seals and the rupture discs will preclude leakage of waste gas to the environment during normal operation of the waste gas disposal system or in the event of unauthorized manual operation of a relief valve.

Effluent from the waste gas decay tanks will be mixed with diluent air and filtered through charcoal filters prior to release to the atmosphere. Discharge will be monitored for flow rate, pressure, temperature, and particulate and gaseous activity prior to release through the process vents. Monitors are set in such a manner that the activity release will result in concentrations at the site boundary less than the 10 CFR

Part 20 limits. In the event that the activity of the effluent stream exceeds the settings of the monitors, valves in the vent line will automatically close, thus terminating the release of radioactive gases from the waste gas decay tanks.

We have reviewed the applicant's evaluation of the release of gaseous radioactivity from the gaseous waste disposal system. The applicant has assumed that both units have operated throughout an entire year with a design basis activity equivalent to that associated with 1% failed fuel in the primary coolant system. This would result in the discharge of approximately 17,500 curies per year, predominantly krypton-85 (approximately 84% of the total) and xenon-133 (approximately 16% of the total). This would result in a whole body dose to an individual at the site boundary of approximately 7 millirem per year. On the basis of our review, we have concluded that the gaseous waste disposal system should permit gaseous effluent releases to be held at a small fraction of the 10 CFR Part 20 limits. The Technical Specifications require that the system be operated in such a manner that releases are as low as practicable.

3.1.8.4 Solid Waste Disposal System

Solid wastes will include concentrated wastes from the waste disposal evaporator, concentrated boric acid not to be reused in the system from the boron recovery bottoms tank, spent resin slurries, spent filter cartridges, and other miscellaneous solid materials resulting from station operation and maintenance. The solid waste disposal system provides holdup, packaging, and storage facilities for solid wastes. Loaded solid waste containers will be monitored and stored in an appropriate area prior to shipment offsite for disposal. No solid wastes will be stored permanently at the Surry site. All solid radioactive wastes ultimately will be shipped offsite for disposal at an AEC-licensed disposal site.

3.1.9 Radiological Consequences of Postulated Accidents

3.1.9.1 General

The applicant has analyzed the consequences of various plant transients which might result from anticipated occurrences during the life of the plant. In all cases, the applicant has concluded that the minimum departure from nucleate boiling (DNB) ratio reached will be greater than 1.30, and that fuel damage will not occur. The applicant has presented

the assumptions, a summary of the analytical methods, and the conclusions of his analyses. We have reviewed the applicant's analyses and find them acceptable.

We have also evaluated the consequences of occurrences that are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. Within this category we have selected five highly unlikely accidents (design basis accidents). We performed conservative analyses of these design basis accidents to assess the capability of the engineered safety features to control the possible escape of fission products from the facility. The design basis accidents analyzed were: (1) loss-of-coolant, (2) fuel handling, (3) steam-line break, (4) steam generator tube rupture, and (5) gas decay tank rupture.

Our evaluation of these accidents showed that the resulting calculated doses would be within the 10 CFR Part 100 guideline values. The results of our analyses are summarized in Table 3.1.9, and the calculation models and assumptions used in each analysis are described in the following sections.

3.1.9.2 Loss-of-Coolant Accident

To control and to minimize the release of fission products to the environment in the unlikely event of a design basis loss-of-coolant accident, Surry Power Station Units 1 and 2 are equipped with reinforced concrete containment structures normally maintained at a subatmospheric pressure. Redundant containment spray systems and containment recirculation spray systems are provided to remove heat generated from the accident and thereby assure that the containment design pressure is not exceeded and that the pressure in the containment following an accident will rapidly decrease to below atmospheric pressure, thus terminating leakage. The most severe accident considered in our evaluation is a postulated double-ended rupture in the largest pipe of the reactor coolant system.

In our evaluation, we have assumed the rupture of a cold-leg pipe attached to the reactor vessel, resulting in a complete loss of reactor coolant. In calculating the potential consequences of this postulated loss-of-coolant accident, to provide a conservative assessment we have arbitrarily assumed that in spite of the operation of the emergency core cooling system, large amounts of fission products would be released from the reactor fuel. The fractions of the total core

fission product inventory we assumed to be released from the core are those given in AEC Safety Guide 4 (100% of the noble gases and 50% of the halogens). As discussed in Section 3.2.2.3 of this evaluation, we have assumed that the containment atmosphere is reduced to below atmospheric pressure within 60 minutes after the occurrence of the loss-of-coolant accident, thus terminating leakage and that leakage exists at a rate of 0.1% of the containment volume per day until that time. Other assumptions are presented in Appendix I. The calculated doses for this postulated accident at the exclusion radius and at the low population zone distance are presented in Table 3.1 and are within the guideline values of 10 CFR Part 100.

3.1.9.3 Refueling Accident

In our evaluation of the refueling accident we have conservatively assumed that during fuel handling operations a dropped fuel bundle containing 204 fuel rods falls with sufficient force to damage all of the fuel rods in the bundle with consequent release of 10% of the noble gases and 10% of the iodines in these rods to the pool water. Other assumptions we used in calculating the potential offsite doses are given in Appendix I. The resultant calculated doses are given in Table 3.1 and are within the guidelines of 10 CFR Part 100.

3.1.9.4 Gas Decay Tank Rupture

We have evaluated the potential radiological consequences of a postulated sudden rupture of one of the two waste gas storage tanks of the gaseous waste treatment system. The highest inventory of waste gases in the tanks occurs at the end of the feed cycle of the gaseous waste disposal system with the recombiner operating. We have analyzed this event assuming the loss of the gases contained in one tank at this time. Other assumptions used in the calculation are given in Appendix I. The resultant calculated doses are given in Table 3.1 and are within the guideline values of 10 CFR Part 100.

3.1.9.5 Steamline Break Accident

If the steamline break is postulated to occur outside containment and downstream of the steamline isolation valve, the break can be isolated and the potential radiological consequences of the accident are minor. If the break occurs between the steamline containment penetration and the isolation valve, the break cannot be isolated and the affected steam generator will discharge steam to the atmosphere. Analyses have indicated that no fuel damage will result from a break of a steamline. However, because of the possibility of reactor coolant leakage to the secondary coolant through steam

generator tube defects, fission product activity entrained in the reactor coolant could be released to the atmosphere. We have calculated the radiological consequences of such a steam-line break accident using the assumptions stated in Appendix I. As shown in Table 3.1, the calculated doses are within the guideline values of 10 CFR Part 100.

3.1.9.6 Steam Generator Tube Rupture Accident

In the event of a sudden rupture of a steam generator tube, radioactive primary coolant would flow into the steam system. The potential consequences depend upon the size of the assumed failure and on other conditions assumed to exist, such as the availability of offsite power.

If offsite power is assumed to be available following a tube rupture, the steam supply to the turbine will be diverted through the condenser bypass valves to the condensers. Any gaseous fission products that leak to the secondary system would be drawn from the condenser by the air ejector. In the event of a high activity alarm from the air ejector, air ejector effluent is automatically diverted to the containment structure and is not released to the atmosphere.

If it is assumed that offsite power is lost concurrent with a tube rupture, the circulating water pumps stop and the condenser and air ejectors are isolated. Decay heat removal is accomplished by boiloff of emergency feedwater in the steam generators, with subsequent release of the steam produced to the atmosphere through the steam generator safety and/or relief valves. During this period, the steam generator experiencing the ruptured tube can be identified through the use of the steam generator blowdown monitor and observation of steam generator levels. Once the faulted steam generator is identified, it can be isolated using the primary system loop isolation valves, terminating leakage in a short period of time.

In our calculation of the offsite doses for this postulated event, we assumed 30 minutes elapsed before the affected steam generator can be isolated. Other assumptions are presented in Appendix I. The calculated offsite doses are within the guideline values of 10 CFR Part 100 as shown in Table 3.1.9.

3.1.10 Conclusions

On the basis of our evaluation of the important site characteristics, and taking into consideration the design of

TABLE 3.1.9

DESIGN BASIS ACCIDENTS
CALCULATED OFFSITE DOSE

Accident	Exclusion Radius (0.31 miles) Thyroid (rem)	Exclusion Radius (0.31 miles) Whole Body (rem)	Low Population Zone (3.0 miles) Thyroid (rem)	Low Population Zone (3.0 miles) Whole Body (rem)
LOSS-OF-COOLANT	220 rem	7 rem	20 rem	<1
FUEL HANDLING	34	3.0	3.0	<1
GAS DECAY TANK RUPTURE	negligible	10	negligible	<1
STEAMLINE BREAK	8	<1	<1	<1
STEAM GENERATOR TUBE RUPTURE	28	1	2.5	<1

the facility including the engineered safety features and the systems provided for the control of radiological effluents from the plant, we have concluded that the site characteristics are in accordance with the Commission's siting guidelines, 10 CFR Part 100 and, therefore, the site of the Surry Power Station is acceptable.

3.2 Facility Engineering and Architectural Features

3.2.1 Nuclear Steam Supply System Design

3.2.1.1 General

The reactor cores for Surry Units 1 and 2 are similar to that previously approved for the H. B. Robinson Unit No. 2. They each consist of 157 fuel assemblies with 204 fuel rods per assembly. The active heat transfer surface area available is the same for these units. The proposed total heat output for the Surry units is 2441 MWt as compared to a total core heat output of 2200 MWt for H. B. Robinson Unit No. 2.

3.2.1.2 Nuclear Design

The design nuclear heat flux hot channel peaking factor is 2.72 while the nuclear enthalpy rise hot channel factor is 1.58. These are less than those approved for the H. B. Robinson plant but identical to the values approved for Point Beach Unit No. 1, which has a smaller core.

Reactor control will be accomplished by means of 48 control rods (16 of which are shutdown rods) and 5 part-length control rod assemblies for control of xenon oscillations. In addition, burnable poison rods will be used in the initial core loading and soluble boron will be used for the control of long-term reactivity changes. This differs from previous Westinghouse 3-loop plants where 45 control rods and 8 part-length rods are provided. The applicant has stated that a local reactivity worth of 0.7% is adequate for control of axial perturbations. The local reactivity worth of the 5 part-length control rods is 1%. The initial startup testing program will verify that the 5 part-length rods are adequate to control xenon oscillations caused by axial power perturbations.

3.2.1.3 Thermal and Hydraulic Design

Each of the Surry units will have an 11% increase in power over H. B. Robinson Unit No. 2. This has been accomplished through the use of lower peaking factors as discussed in Section 3.2.1.2, above. Use of the lower peaking factors results in a minimum DNB ratio of 1.97 at nominal conditions and 1.30 for design transients. A comparison of the thermal and hydraulic design parameters for Surry and H. B. Robinson Unit 2 is presented in Table 3.2.

As discussed in Amendment No. 25 to the application, the applicant proposed a change in the allowable peak linear power density limits. This change was proposed to achieve conformance with the acceptance criteria setforth in the Commission's Interim Policy Statement on emergency core cooling systems. Our evaluation of the performance of the emergency core cooling systems is presented in section 3.2.2.2.

The applicant has proposed to limit the peak linear power density to 16.1 kilowatts per foot at 100% of rated power. This requires a further reduction in the peaking factor from 2.72 to 2.6. As summarized in Amendment No. 25 to the application, a large body of data from currently operating reactors is available to support the contention that the lower total peaking factor of 2.6 can be achieved by imposing operating restrictions. Measured peaking factors at six different plants reported in Amendment No. 25 indicate that operating plants, without additional restrictions, presently operate at total peaking factors well below 2.6.

We have concluded that the thermal and hydraulic design of the Surry plant is acceptable.

TABLE 3.2

COMPARISON OF THERMAL AND HYDRAULIC DESIGNS OF SURRY AND ROBINSON

Parameter	Surry	H. B. Robinson 2
Thermal Power (MWt)	2441	2200
Nuclear Heat Flux Hot Channel Factor	2.72 (2.60)*	3.13
Enthalpy Rise Nuclear Hot Channel Factor	1.58 (1.50)*	1.73
Heat Transfer Surface ($feet^2$)	42,460	42,460
Average Heat Flux ($Btu/hr/ft^2$)	191,100	171,600
Maximum Heat Flux ($Btu/hr/ft^2$)	534,100 (510,000)*	554,200
Average Specific Power at 100% Power (kW/ft)	6.2	5.5
DNBR at Nominal Conditions	1.97	1.81
Minimum DMBR for Design Transients	1.30	1.30

*Interim limits

3.2.1.4 Mechanical Design

We have reviewed the mechanical design of the reactor internals, which are substantially the same as those installed in some previously reviewed plants. We have concluded that the methods used to implement the approved design criteria and the analytical methods employed to determine the effects of postulated accidents, the design basis earthquake, and the loads which would result from the concurrent occurrence of these events, have resulted in acceptable components. Thus, we conclude that the reactor internals are capable of withstanding the loads which would result from the simultaneous occurrence of the design basis earthquake and the loss-of-coolant accident and are acceptable.

Since no significant design or fabrication differences exist between the Surry and Robinson units that could significantly affect internals vibration characteristics, no vibration monitoring program, other than the visual inspection of the reactor internals, is planned during initial startup. We have reviewed the experimental results obtained from the internals vibrations tests performed at the H. B. Robinson facility and have determined that they support the analytical method used to analyze the Surry internals. On this basis, we have concluded that a vibration test program at Surry is not

warranted and that the analytical method of determining core internals vibration characteristics coupled with visual inspection is appropriate.

3.2.1.5 Reactor Coolant System

The primary coolant system consists of three heat transfer loops, each of which is equipped with two loop stop valves to permit isolation of a steam generator and reactor coolant pump. The reactor vessel for Unit 1 was designed and partially fabricated by the Babcock & Wilcox Company; subsequently it was transferred to the Rotterdam Dock Yard Company (RDM) for completion of fabrication. The vessel is designed as a Class A vessel to the requirements of Section III of the ASME Boiler and Pressure Vessel Code. Work done at the Babcock & Wilcox Company was based on the 1965 edition of the aforementioned Code and all addenda up to Summer 1966. The RDM work was based on the 1968 edition of the Code and all addenda up to Winter 1968.

An AEC inspection team visited RDM in 1969, and concluded that the manufacturing organization, facilities, and practices at RDM provide reasonable assurance that nuclear vessel components produced there meet quality levels comparable to those achieved for vessels manufactured in the United States. Subsequent inspections have supported this conclusion.

We have concluded that the RDM facilities and qualifications are equivalent to those of U. S. manufacturers and that appropriate design codes were used for the Surry vessels in accordance with the requirements of Section 55.55a, 10 CFR Part 50.

Primary system leak detection is accomplished by the use of containment atmosphere radioactivity monitors and the level and flow indicators in the reactor sump. Indirect indication can also be obtained from containment pressure, temperature, and humidity detectors. The system provided at Surry is substantially the same as systems approved for other recently reviewed facilities. On this basis, we have concluded that the leakage detection system for Surry is adequate to detect small leaks in the reactor pressure boundary in a timely manner and is acceptable.

An inservice inspection program for the reactor coolant pressure boundary has been proposed by the applicant. We have reviewed the program proposed and have determined that it satisfies the provisions of the ASME Boiler and Pressure Vessel Code, Section XI, and is consistent with those approved for other recently reviewed facilities. On this basis, we

have concluded that the proposed program for Surry is acceptable, and have incorporated it into the Technical Specifications for the facility.

3.2.1.6 Stainless Steel Weldments Containing Microfissures

Investigations at the site during construction revealed low ferrite content in essentially all of the heavy section type 316 stainless steel weldments and in a limited number of weldments of smaller section thicknesses in type 304 stainless steel piping. Examination of a number of specimens removed from the piping revealed that the specimens contained a number of microfissures. The applicant, through its contractor, Southwest Research Institute, conducted a comprehensive laboratory investigation to determine what effect, if any, the low ferrite level and microfissures might have on the integrity of the welds. Approximately 100 boat samples from 25 completed welds in the main coolant loops and nine completed weld sections from safety related auxiliary systems were removed and examined in a metallurgical and mechanical test program. This program included determinations of fatigue, tensile, and chemical properties of the materials and microfissure sizes, densities, and distribution. The test program indicated that microfissures of small size and occupying only a small fraction of the load carrying area of the weld, as observed at

Surry, have no significant effect on the mechanical properties of the material. Further, an analytical evaluation, using fracture mechanics techniques, supported the results of the test program.

On the basis of the test program and analytical calculations performed by the applicant, and results of an independent test program performed for the AEC by Battelle Memorial Institute, we have concluded that the safety of the Surry facility is not jeopardized by the presence of microfissures.

3.2.2 Engineered Safety Features

3.2.2.1 General

Engineered safety features are provided to mitigate the consequences of the postulated accidents discussed in Section 3.1.9. These include the emergency core cooling system, the containment spray system, the post-loss-of-coolant accident combustible gas control system, and the subatmospheric containment. Each of these systems is discussed below.

3.2.2.2 Emergency Core Cooling System (ECCS)

The emergency core cooling system is designed to provide emergency core cooling for those postulated accidents where it is assumed that a failure in the reactor coolant system results in a loss of coolant from the system greater than the makeup capacity of the charging pumps. Sufficient redundancy is provided so that no single failure of emergency core cooling system equipment occurring during a loss-of-coolant accident will result in inadequate cooling of the reactor core.

The Surry emergency core cooling system consists of a high-head safety injection system, a low-head safety injection system, and an accumulator injection system. The high-head system utilizes the charging pumps and takes suction from either the refueling water storage tank or the discharge of the low-head safety injection pumps, after isolating the normal charging lines. The charging pumps, acting as high-head safety injection pumps, normally discharge through the boron injection tank to the three cold legs. The low-head safety injection pumps take suction from either the refueling water storage tank or the containment sump and normally discharge to the three reactor coolant system cold legs. The safety injection will be transferred from cold-leg to hot-leg injection using motor-operated valves after the termination of the temperature transient in order to suppress boiling in the core region. However, to preclude inadvertent operation of these valves during the early stages of the operation of the emergency core cooling system following a loss of coolant accident, a key-lock switch activated from the control room will be installed prior to power operation.

The accumulators discharge to the three cold legs and are isolated from the reactor coolant system cold legs by two

check valves and a normally open gate valve. Because an inadvertent closing of any one or more of these gate valves due to an administrative error or an electrical fault concurrent with the requirement for safety injection could seriously impair the ability of the emergency core cooling system to adequately cool the core, the applicant has provided visual indication of the position of the gate valves in the control room. The indication is actuated by sensors on the valve for the open and closed positions. In addition, an audible alarm which is actuated by a sensor on the valve when the valve is not in the fully open position has been installed. The safety injection system signal which actuates the pumps in the emergency core cooling system will override any by-pass feature that may be provided to allow a motor-operated valve to be closed when the primary system is at pressure and will initiate valve opening if the valve is closed. Prior to the return to power following the first refueling of Surry Unit No. 1, the applicant will incorporate automatic opening of the gate valves whenever the reactor coolant system pressure exceeds a preselected value. In the interim period, power to the valve operator will be locked out during power operation with the valve in the open position.

All of the emergency core cooling subsystems will be designed to accomplish their functions when operating on emergency (onsite) power as well as offsite power. In the event of a loss of offsite power concurrent with a single failure in the emergency power supply system, the minimum emergency core cooling system requirement of the accumulators (which require no electrical power), plus one charging pump and one low head safety injection pump would be available for operation and capable of providing the required performance.

Technology associated with analyses of ECCS performance capability has developed substantially in the past five years. As a consequence of this expanding technology, improvements in analytical techniques have been evolving. Recently, additional changes and improvements have been made in these analytical techniques for use in analyses of loss-of-coolant accidents. As a result of these changes and because of the results of some preliminary safety research experiments, we have reevaluated the effectiveness of emergency core cooling systems in light-water cooled reactors.

On June 29, 1971, the AEC issued an Interim Policy Statement[1/] containing interim acceptance criteria for the performance of emergency core cooling systems for light-water cooled nuclear power reactors. A public rule making hearing on the Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors is currently in process. The Interim Policy Statement includes a set of conservative assumptions and procedures to be used in analyzing the ECCS functions for pressurized water reactor plants incorporating a dry containment.

In Amendment No. 25 to the application, the applicant provided the results of an analysis of the ECCS performance capability for the Surry power station. We have reviewed this analysis and have concluded that it was performed using the Westinghouse evaluation model described in Appendix A, Part 3 of the Interim Policy Statement with certain modifications (as described below) to account for differences between a subatmospheric containment system and the standard dry containment.

Item 7, Appendix A, Part 3 of the Interim Policy Statement requires that the containment back pressure assumed for the analyses of reflooding heat tranfer should not be higher

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than the initial pre-break pressure plus 80% of the increase in pressure calculated for the accident under consideration. In view of the fact that containment pressure decreases more rapidly in a subatmospheric containment than in a normal dry containment (as discussed in Section 3.2.2.3, below), the applicant assumed a reflooding back pressure of 40 psia, equivalent to the initial pre-break pressure plus approximately 70% of the increase in pressure calculated for the accident under consideration. We have evaluated the applicant's assumption in this regard and have reviewed the sensitivity studies performed by the applicant on the effect of reduced containment back pressure. Based on this review, we have concluded that the deviation in the calculational technique from that stated in the evaluation model represents a conservatively low estimate of the pressure transient and it is acceptable. In all other regards, the calculations performed by the applicant conform with the evaluation model stated in the Interim Policy Statement and are acceptable.

To meet the acceptance criterion of the Interim Policy Statement limiting the peak clad temperature to less than 2300°F, the applicant proposes to reduce the maximum allowable

total peaking factor from 2.72 to 2.60 (approximately 7%). As discussed in Section 3.2.1.3, we find this peaking factor reduction to be acceptable.

In Amendment No. 25, the applicant presented the results of analyses of calculated peak clad temperatures for a spectrum of pipe break sizes up to and including the double ended rupture of the largest coolant pipe. The calculated peak clad temperatures, assuming plant operation at 102% of the design power level of 2441 MWt, are as follows:

Break Description	Peak Clad Temperature	Hot Spot Metal-Water Reaction
Double-ended cold leg (Guillotine) ($8.24\ Ft^2$)	2300°F	8.6%
Double-ended cold leg (Guillotine) ($6.59\ Ft^2$)	2290°F	8.5%
Cold-leg split ($4.5\ Ft^2$)	2240°F	7.4%
Cold-leg split ($3.0\ Ft^2$)	2014°F	3.2%
Cold-leg split ($0.5\ Ft^2$)	1860°F	1.5%

The results of these analyses also indicate that for each of the assumed pipe breaks, the total core metal-water reaction is less than 1%. The clad temperature remains at an

elevated temperature for only a short period of time so that no significant amount of cladding would become embrittled and the core geometry would remain amenable to the long-term removal of decay heat by the emergency core cooling system.

As indicated in Section 3.2.1.4, we have concluded that the reactor internals are capable of withstanding the loads which would result from the simultaneous occurrence of the design basis earthquake and the loss-of-coolant accident. Thus, blowdown forces will not result in distortions which could prevent core cooling.

The emergency core cooling system continues to remove decay heat and to reduce the core temperature following a transient. Emergency core cooling water which spills from the reactor coolant system rupture collects in the containment sump where it is mixed with the cooler containment spray and recirculation spray water. The sump water is recirculated to the reactor vessel by the low head safety injection system. No single failure of active emergency core cooling system components, and no single failure of passive components during the long-term cooling phase will reduce system performance below acceptable levels.

On the basis of our evaluation, we have concluded that the predicted functional performance of the Surry emergency core cooling system for the full spectrum of postulated break sizes up to and including a double-ended break in the largest cooling pipe is in accord with the Commission's Interim Policy Statement of acceptance criteria and is acceptable.

3.2.2.3 Subatmospheric Containment Concept

The applicant proposes to operate the containment at a pressure of approximately 10 psia. Following a loss-of-coolant accident, the containment pressure would increase and subsequently decay in a manner similar to that of other dry containment systems. However, the containment spray system using water chilled to approximately 45°F together with a recirculation spray system which draws water from the containment sump and pumps it though heat exchangers to spray headers within containment are provided to rapidly depressurize the containment structure to below atmospheric pressure following a loss-of-coolant accident. The applicant has determined that the peak containment pressure reached following a loss-of-coolant accident is 39.2 psig, and that the containment returns to subatmospheric conditions 38 minutes after the occurrence of the accident. We have independently calculated a peak containment pressure of 41.3 psig, using the

CONTEMPT code. We have evaluated the results and have concluded that adequate margin exists between the calculated peak pressure and the containment design pressure of 45 psig.

We have reviewed the evaluation methods used by the applicant and his contractors for establishing the containment response to accidents and have concluded that the methods are acceptable.

We have analyzed the consequences of the loss-of-coolant accident presented in Section 3.1.9.2 of this evaluation assuming the containment leaks at its design leakage rate of 0.1% per day for a period of 60 minutes following a loss-of-coolant accident, and that no out-leakage occurs thereafter. Based on our evaluation of the analytical techniques used by the applicant to calculate depressurization time, we have concluded that the increase in depressurization time from the 38 minutes calculated by the applicant to the 60 minutes used in the staff analysis represents a conservative estimate of the maximum length of time out-leakage could occur.

3.2.2.4 Containment Spray

Containment sprays are provided to cool the containment and return it to a subatmospheric condition following a loss-of-coolant accident. Two subsystems are provided. The first,

identified as the containment spray system, draws water from the refueling water storage tank and sprays it into the containment; the second, identified as the recirculation spray system, draws water from the containment sump, cools it, and again sprays it into the containment.

The containment spray systems supply chilled water (approximately 45°F) from the refueling water storage tank to the containment immediately following receipt of an engineered safety feature initiation signal. The system consists of two independent trains of equipment, each consisting of a spray pump equipped with a dual steam turbine and electric motor drive, and a 360° containment spray header. Either train is capable of furnishing sufficient spray water to reduce the pressure in the containment structure to below atmospheric pressure when operating in conjunction with the recirculation spray system. A chemical addition tank containing concentrated sodium hydroxide is balanced hydraulically with the refueling water storage tank and provides a flow of sodium hydroxide solution to increase the capability of the containment spray to remove iodine.

The recirculation spray system consists of four independent trains of equipment each consisting of a recirculation

spray pump, a heat exchanger, and a 180° recirculation spray header. Two of the four 50% design capacity motor-driven recirculation spray pumps are located outside the containment on an extension of the containment mat. The other two trains are located entirely within containment. Two trains of the recirculation spray subsystem operating in conjunction with one train of the containment spray subsystem are capable of reducing the pressure in the containment structure to below atmospheric pressure following a loss-of-coolant accident.

We have reviewed the design of the system as installed and the redundancy provided for vital equipment and have concluded that the system is adequate to cool the containment following a loss-of-coolant accident.

We have also evaluated the iodine removal characteristics of the spray system using the measured drop height and size distribution supplied by the applicant, and system parameters. We have determined that an iodine removal constant (λ) of 2.2 hr^{-1} is appropriate, yielding a 1-hour iodine dose reduction factor of 2.0.

3.2.2.5 Containment Combustible Gas Control

Following a loss-of-coolant accident, hydrogen would be generated inside the containment from (1) a chemical reaction between the fuel rod cladding and the steam resulting from vaporization of emergency core cooling system water (metal-water reaction), (2) corrosion of metals inside containment by the spray solution, and (3) radiolytic decomposition of recirculating coolant solutions, with concurrent oxygen generation. If a sufficient amount of hydrogen is generated, its subsequent reaction with oxygen at rates rapid enough to lead to a significant over-pressure could lead to degradation of the ability of the containment to maintain its low leakage integrity.

The applicant has calculated that the hydrogen concentration in the containment (without operation of recombiners) will reach 4% in 13 days assuming a 2% metal-water reaction, and in 8 days assuming a 5% metal-water reaction. Using the somewhat more conservative assumptions stated in AEC Safety Guide No. 7, we have calculated that a 4% hydrogen concentration will be reached in 6.4 days. To maintain the hydrogen concentration at this level in the absence of control equipment, we have calculated that a containment purge rate of approximately 53,000 cubic feet per day (37

scfm) is required. A purge system capable of providing this purge rate and provided with appropriate filters will be installed and be operable prior to plant operation. We calculate doses resulting from purging as 0.6 rem whole body and 14 rem to the thyroid at the outer boundary of the low population zone. These doses are for the entire post-accident period and assume meteorological conditions equivalent to those specified for 4-30 day meteorlogical conditions in AEC Safety Guide No. 4. When combined with the calculated doses associated with a loss-of-coolant accident presented in Table 3.1, the total offsite doses are still within the guidelines of 10 CFR Part 100.

The applicant has agreed to install means of limiting the hydrogen concentration in the containment following a loss-of-coolant accident to below the lower flammability limit of hydrogen in oxygen (approximately 4%) by the end of 1972. To accomplish this, two 100% capacity catalytic hydrogen-oxygen recombiners will be installed, each having a minimum flow rate of 50 standard cubic feet per minute (scfm). In addition to the two recombiners, the purging capability discussed above will be available as a backup.

Convective mixing associated with operation of the containment spray system will be relied upon to assure a

uniform mixture of hydrogen within the containment. No fans are incorporated within the plant design that will be operable following a loss-of-coolant accident. We have examined the experimental information on mixing caused by containment spray systems obtained from the Containment System Experiment tests. Considering these data and the location and size of openings venting the internal compartments of the containment, we have concluded that adequate mixing will be obtained.

3.2.3 Design Bases for Equipment and Structures

The applicant classified the facility equipment and structures into two categories according to function or consequence of failure. The Class I (seismic) category includes structures and equipment whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of excessive amounts of radioactivity, or that are vital to safe shutdown and isolation of the reactor. The Class II (seismic) category includes structures and equipment defined as those that are not related to reactor operation or public safety and are not Class I (seismic). We have reviewed the applicant's classification of structures and equipment and have concluded they have been appropriately classified.

The reactor coolant pressure boundary components and component support designs have been analyzed to check their capability to withstand the stresses associated with the combined loadings of the design basis loss-of-coolant accident and the design basis earthquake. The results of these analyses indicate that some permanent deformation might occur, but that it would be well below an average of 20% of the uniform ultimate strain for the materials of construction and would not result in a loss of function. The design, fabrication, examination, and test criteria used for Surry are equivalent to or provide greater conservatism than those accepted for all recent plants of this type. We have concluded that the use of these criteria has resulted in acceptable designs for the components of the reactor coolant pressure boundary.

We have evaluated the codes and standards specified for the Class I (seismic) tanks, heat exchangers, piping, pumps and valves. We have concluded that the codes and standards used for Surry were consistent with those used for recently reviewed plants of this type and that their use resulted in an acceptable quality level. All Class I (seismic) items are designed to withstand the operational basis earthquake within appropriate code allowable stress limits and the design basis

earthquake within stress limits comparable to those associated with the emergency operating condition category of current component codes. We consider that the use of such stress criteria provides an adequate margin of safety for Class I (seismic) systems and components.

The modal response spectrum method of analysis was used for design of Class I (seismic) structures, systems and components. We and our seismic design consultant, Nathan M. Newmark Consulting Engineering Services, have reviewed the method of analysis and have concluded that it provides an adequate basis for acceptance of the approach used in the design of the Surry components and structures. The report of Nathan M. Newmark Consulting Engineering Services is attached as Appendix G. The applicant has reviewed the seismic analyses adding factors to account for vertical and horizontal amplified structural response. Seismic stresses combined with normal loadings are within the limits for the emergency operating conditions category defined in Case 70 of Interpretations of Code for Pressure Piping. On the basis of the above, we find the seismic design of Class I (seismic) structures, systems, and components to be acceptable.

The reactor containment structure consists of a steel-lined reinforced concrete cylinder and hemispherical dome

supported on a base mat on soil at 66 feet below finished grade. It is similar in design to the Connecticut Yankee containment structure. The containment design incorporates reinforcing bars at 45° in the wall to resist tangential seismic shears in addition to the vertical and hoop reinforcing. The design load combinations and the analyses and design criteria used in considering these loads for the Surry containment have been reviewed and found acceptable.

The containment structure was analyzed for seismic loads using a computer program with selected results confirmed by hand computation techniques. The design criteria, the design loads and the results of the analyses of the structural design and seismic response of the Surry Containment structures have been reviewed and found to be acceptable.

Transient pressure conditions in the internal compartments of the containment have been calculated by the applicant. The steam generator cubicles and the pressurizer cubicle are designed for a differential pressure of 35 psi. The primary shield which forms the reactor vessel cavity is designed to withstand an internal pressure of 100 psi. The

applicant has stated that a peak reactor vessel cavity differential pressure of approximately 61 psi would result from a 4.6 square foot rupture of a reactor coolant pipe (equivalent to a single-ended rupture).

The containment structure will be tested for structural integrity by pressurizing it with air to 115 percent of design pressure, then held at 115% of design pressure for 48 hours. Visual and instrument observations of the structure will be conducted during the test. Due to the materials used, the stress levels involved, and the structural configurations, no in-service structural surveillance tests are planned. However, the initial structural acceptance test could be duplicated at any time further tests were required. Periodic integrated leakage rate tests will be conducted at a frequency specified in the Technical Specifications.

3.2.4 Instrumentation, Control and Electrical Power Systems

3.2.4.1 General

Our review of the Surry plant instrumentation, control, and electrical power systems was directed at those subsystems and features that are related to the safety of the plant. Particular attention was given to the review of the reactor protection system, the initiating and control systems for engineered safety features, and the emergency power portions

of the plant auxiliary electrical power supply. The Commission's General Design Criteria outlined in Appendix A to 10 CFR Part 50, and the Institute of Electrical and Electronic Engineers Criteria for Nuclear Power Plant Protection Systems (IEEE-279) served, where applicable, as the bases for evaluating the adequacy of the design, as installed.

The reactor trip system is substantially the same as that installed at H. B. Robinson Unit 2 and previously reviewed by the staff, while the engineered safety feature initiating circuits are substantially different. Consequently, our review of the reactor trip system concentrated on those items which are unique to this station, or for which new information has been received. The engineered safety feature initiating circuitry was examined in detail. No deficiencies were observed in the design of the reactor trip, safety injection, steamline isolation, emergency feedwater, containment spray, containment isolation, and loop stop valve interlocks. The applicant's cable installations have been reviewed with regard to (1) the preservation of independance of redundant channels by means of separation and (2) prevention of cable fires through derating and proper tray loading, and found to be acceptable.

The environmental tests performed on equipment that must operate in the event of (1) a seismic event or (2) a loss-of-coolant accident have been reviewed and found to be acceptable with the exception of the environmental tests performed on the recirculation spray pump motors located inside containment. A pre-aged, pre-irradiated prototype of a containment recirculation spray pump motor was tested under simulated loss-of-coolant accident conditions. During the test, failures occurred which have led us to the conclusion that these test results do not provide adequate bases for judging the motor acceptable for its intended service. In Amendment No. 28 to the application, the applicant stated he will review additional test data from tests conducted at other facilities and will either resubmit an expanded test report on the existing motor, retest the existing motor, or replace the motors with other acceptable motors. We will not permit operation above 50% of rated power until the containment recirculation spray pump motors located inside containment have been adequately tested and found to be acceptable. At power levels below 50% of rated power, adequate redundancy is provided by the two recirculation spray pumps located outside containment which would not be subjected to the post-loss-of-coolant accident environment.

3.2.4.2 Electrical Power Systems

The Surry Power Station contains two substations, a 230 kV substation and a 500 kV substation which are interconnected and mutually redundant. A switchyard battery is assigned to each substation to provide breaker control power. The 230 kV transmission lines leading to the 230 kV substation follow two independent rights-of-way. In addition, there is a third independent right-of-way for the 500 kV lines. The two startup transformers are fed from separate buses. We have reviewed the design, as installed, and have concluded that it is acceptable.

The onsite emergency power system is shared between Surry Unit No. 1 and Surry Unit No. 2. Two emergency buses are provided for each unit, and are served by three diesel generators arranged in a split-bus configuration, with one bus in each unit served by its own diesel and the two remaining buses served by a shared (or swing) diesel. Each diesel is automatically started by under voltage relays at its respective bus and by safety injection and high containment pressure signals originating in the engineered safety feature train to which the diesel and its bus are assigned. The

starting systems are independent of each other. Each diesel has its own set of fuel oil transfer pumps. The valve control circuits for the fuel oil transfer system have been reviewed and found to satisfy the single failure criterion and, thus, to be acceptable. Diesel loads are limited by the Technical Specifications to conform to the regulatory position stated in AEC Safety Guide No. 9.

Four station batteries are provided (two for each unit) and are respectively assigned to the four emergency buses. They are located in two adjacent battery rooms. DC cables run separately in conduit from each room and terminate in physically separate switchgear located outside the battery room. We have concluded that the design of the dc system, including the seismic design of the battery racks is acceptable.

We have reviewed the onsite emergency power system and have determined that the design, as installed, satisfies the single failure criterion, and that adequate separation is provided. Therefore, we conclude that the emergency power system meets the requirements of IEEE-308 and conforms with the regulatory position stated in AEC Safety Guides Nos. 6 and 9 and is acceptable.

3.2.4.3 Anticipated Transients Without Scram

The ACRS has recommended that the applicant expedite the studies of further means of preventing common mode failures from negating reactor scram action, and design features to make tolerable the consequences of failure to scram during anticipated transients. Westinghouse has performed several analyses in this regard which are presently under review by the regulatory staff. If it is determined that design modifications could significantly improve plant safety, they will be required at Surry on a timely basis. In the interim period, we have concluded on the bases of the analyses performed and evaluated that the system as proposed is acceptable.

3.2.5 Auxiliary Systems

3.2.5.1 General

The auxiliary systems are described in Section 9 of the Final Safety Analysis Report. These process systems normally provide plant services that have an auxiliary function to the production of power. In the course of our review, we have directed our attention to the safety related design bases of these systems and the manner in which these design bases were

implemented. We have reviewed the design criteria including redundancy, independence, physical separation, and those criteria that established the quality of the system. The latter review considered the appropriateness of the seismic design classification, and the use of suitable codes, standards, and specifications for the design, fabrication and inspection of the piping and other components within the system. Those auxiliary systems that received special attention in the course of our review are discussed in the following sections.

3.2.5.2 Circulating Water and Service Water Systems

The circulating water system provides cooling water for the main condensers and for the service water systems of both units. The circulating water intake structure is located at the east end of the site. Eight circulating water pumps, each rated at 210,000 gpm at a total dynamic head of 28 feet, pump river water through 96-inch diameter steel pipes that convey the water over the embankment of and into the high level intake canal. The high level intake canal is approximately 1.7 miles long and is paved with 4-1/2 inches of reinforced

concrete. Circulating water flows by gravity down the canal to the power station, located at the west end of the site. A reinforced concrete high level intake structure is provided in the high level intake canal at each unit. Each structure contains a trash rack, a traveling screen, and an inlet of 96-inch diameter condenser intake line. Since the high level intake canal is designed to Class I (seismic) specifications and is immune to failure from other natural phenomena, we have concluded that this single cooling water source is acceptable.

Service water will be supplied to the recirculation spray heat exchangers from the circulating water system by gravity flow between the high level intake canal and the discharge canal seal pit. During normal operation, the water in the intake canal is approximately 20 feet above the level in the seal pit at the discharge canal. This elevation head provides the driving force for service water flow through the recirculation spray heat exchangers.

In the event of a loss of station power at the river intake, three diesel-driven emergency service water pumps are provided for both units at the river intake structure to supply makeup to the high level canal. Each pump is sized for

a flow of 15,000 gpm and a total head of 45 feet and each can supply sufficient water to provide coolant for engineered safety features following a loss-of-coolant accident or a loss of station power. A 4800 gallon diesel fuel oil storage tank provides sufficient fuel to operate all three pumps for 125 hours. In addition, one diesel-driven pump is equipped with an electric motor drive. The electric motor-driven pump is normally used to maintain the intake canal full of water when both units are shut down. Sufficient water is stored in the canal to provide adequate cooling for 100 hours following an accident without makeup. We have reviewed the system design, as installed, and have concluded that the service water system and the circulating water system are acceptable.

3.2.5.3 Fuel Handling and Storage

Fuel handling and storage facilities will be provided for storage and transfer of new and spent fuel. New fuel will be stored in racks in the new fuel storage area, which does not contain water. The new fuel storage area is sized for storing two-thirds of a core.

Spent fuel will be stored under water in the spent fuel storage pool located adjacent to and between the two reactor containment structures. Subsequently, spent fuel will be transferred to a fuel shipping cask which is submerged in the

fuel pool. After loading, the cask will be removed from the pool and shipped to a fuel reprocessing plant. The fuel building crane is arranged in such a manner that heavy objects, including the spent fuel cask, physically cannot be moved over the stored fuel. A built-up pad of energy-absorbing material is placed over the base of the pit in the area where the cask is loaded. The applicant has stated that if a fuel cask is dropped into the pit, this pad will reduce the impact to a level within the design capability of the reinforced concrete fuel pit structure, so that structural damage to the concrete will not occur.

3.2.5.4 Residual Heat Removal System

In order to cool the primary system to permit depressurization following reactor shutdown, the temperature of the reactor coolant system is initially reduced by transferring heat from the reactor coolant system to the steam system by boil-off in the steam generators. After the primary system is depressurized to below the design pressure of the residual heat removal system (approximately 450 psig), the residual heat removal system is used to remove residual and sensible heat from the core and to reduce the temperature of the reactor coolant system to below 200°F. The residual heat

removal system is not required to operate following a loss-of-coolant accident. The residual heat removal system is located entirely within containment.

The residual heat removal system pumps take suction from the hot leg of one of the three reactor coolant system loops at a point located between the reactor pressure vessel and the loop isolation valve. As originally proposed, the suction line to the residual heat removal system pumps contained two electric motor-operated gate valves in series, the first interlocked with primary coolant system pressure in such a manner that the valve would not open when the primary coolant system pressure exceeded a preselected value. The second valve was manually closed by an administratively controlled switch in the main control room. If these valves were open during power operation, the low design pressure residual heat removal system would be connected to the high pressure design primary coolant system and the residual heat removal system could fail causing a loss-of-coolant accident within containment. The applicant has agreed to modify this system prior to completion of the first refueling to provide pressure interlocks on both valves in the suction line to the residual heat removal pumps which will prevent the valves from opening when

the reactor coolant system is at pressure. In addition, additional interlocks will be provided which will automatically close these valves when the reactor coolant system pressure exceeds 465 psig. During the interim period until the return to power following the first refueling, the power to the valve operators will be locked out during power operation of the reactor with the valve in the closed position, thus preventing inadvertent valve operation from the main control room.

3.3 Inservice Inspection

An inservice inspection program has been developed to provide assurance of the continued integrity of the reactor vessel, the reactor coolant system pressure boundary, and related engineered safety feature equipment throughout the life of the plant. The inservice inspection program proposed is based on Section XI of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, dated January 1970, and the 1970 Addenda. The inspection program was developed by adopting, insofar as practicable, the principles and intent embodied in the Code. The techniques anticipated for inservice inspection include visual inspection, ultrasonic, radiographic, magnetic particle, and liquid penetrant testing of selected parts during refueling periods and necessary maintenance outages. We have reviewed the

inservice inspection program proposed and have determined that the program for the reactor coolant pressure boundary satisfies the provisions of the ASME Boiler and Pressure Vessel Code, Section XI, and is consistent with those approved for other recent reviewed facilities. Further, we have determined that the inservice inspection program proposed for the engineered safety features which are not a part of the reactor coolant pressure boundary are adequate to detect evidence of deterioration or degradation in the quality of the system. On this basis, we have concluded that the proposed program for Surry is acceptable.

3.4 Sensitized Stainless Steel

During construction of the Surry Power Station, certain pieces of cold bent stainless steel piping, used in the reactor coolant system and various safety related auxiliary systems and engineered safety systems became sensitized. Austenitic stainless steel, when heated and held sufficiently long in the 800-1600°F temperature range, or slowly cooled through this range is subject to chromium carbide precipitation at the grain boundaries. In forming this carbide, chromium is depleted from areas adjacent to the carbides. These chromium impoverished areas have a lower corrosion resistance than the steel would normally have. Localized intergranular corrosion can occur in certain corrosive

environments. Material which contains precipitated carbides at the grain boundaries and which has a lower resistance to corrosion than solution annealed stainless steel in a specific environment is said to be sensitized. Sensitization has occurred in the piping of the recirculation spray system, the containment spray system, the low head safety injection system, the high head safety injection system, the charging and letdown system, and in the pressurizer relief lines and in the pressurizer spray and surge lines. The applicant has reheat-treated the pressurizer relief lines to remove the sensitization in these lines.

With respect to the other systems, we have evaluated experimental corrosion data and have determined that (1) chloride stress corrosion of sensitized stainless steel can occur at low concentrations of chlorides in an acid solution but is strongly inhibited in an alkaline solution with a pH of 8.0 or higher and (2) chloride stress corrosion is inhibited by boric acid at a concentration of 2500 ppm.

We have considered each of the systems containing sensitized stainless steel piping and have reached the conclusions stated below:

The recirculation spray system and containment spray system headers will be maintained dry during normal operation.

Following a loss-of-coolant accident, the containment spray system headers will be exposed to 45°F water, containing 2,000 ppm boric acid at a minimum pH of 9.5 for approximately 40 minutes. The recirculation spray system will be exposed to low pH water for a short period of time following a loss-of-coolant accident. However, the pH will reach a value of 8.1 in approximately 40 minutes due to the sodium hydroxide addition to the containment by the containment spray system and will remain basic thereafter. The boric acid concentration in the recirculation spray system will be 2,000 ppm. We have concluded that the containment spray system headers are acceptable as installed because they are exposed only to alkaline borated water. Further, we have concluded that the short term exposure of the recirculation spray system sensitized piping to acidic sump water will not be deleterious to a significant extent and that the system as installed is acceptable.

The pressurizer spray lines and the charging and letdown system lines which are sensitized are not required following a loss-of-coolant accident. During normal operation, the systems will contain flowing reactor coolant system water. The Technical Specifications will limit the fluoride and chloride concentrations each to 0.15 ppm. Because the system will

contain flowing reactor coolant with low concentrations of chlorides and fluorides, we conclude the installed piping is acceptable.

Sensitized steel piping containing low pH borated water solution will be flushed or drained periodically to prevent contaminant buildup or the solution in the lines will be buffered to maintain a basic pH. Further, sampling of the solution in the lines for halogen concentrations will be performed periodically. Following a loss-of-coolant accident, these normally stagnant lines will be exposed to conditions similar to those discussed above for the containment spray system and the recirculation system spray system. Because of the short exposure to acidic borated water following a loss-of-coolant accident and the measures taken to prevent corrosion during normal operation we have concluded that the systems as installed are acceptable.

All other lines containing sensitized stainless steel are isolated from the reactor coolant pressure boundary by at least two isolation valves and are not required for engineered safety feature operation. Thus, failure of these lines would neither cause an accident nor prevent engineered safety feature operation and the lines, as installed, are acceptable.

3.5 Emergency Planning

The applicant has prepared a site emergency plan for dealing with emergencies which might develop during normal operation, such as fires, severe weather conditions, equipment failure and/or uncontrolled release of radioactive materials. During an emergency situation, the shift supervisor will act as the emergency director until relieved by the station manager or his designated alternate. The emergency director will be responsible for making an initial classification of the type of emergency and initiating emergency-controlling actions including alerting station personnel, accounting for all onsite personnel and providing for their safety, contacting members of the emergency committee, isolating unsafe areas, determining the extent of personnel exposure, maintaining a continuous evaluation of the status of the unaffected reactor, and upon verification of the type of emergency, ordering notification of offsite agencies.

An emergency committee consisting of members of the station staff has been established to aid the station manager in verifying the status of the emergency, checking prior actions of the shift supervisor, evaluating, directing, and coordinating onsite activities, dispatching emergency teams offsite, and notifying state and local agencies.

All offsite agencies who potentially will be called on to participate in the emergency plan have been contacted and written agreements exist with regard to their participation. These include a designated physician in Surry County, the Vepco Medical Director in Richmond, local rescue squads, The Medical College of Virginia in Richmond, the State Department of Health and the Office of Civil Defense, the State police, the State Department of Game and Inland Fisheries, the State Forestry Department, and the U.S. Coast Guard.

Prime responsibility for offsite radiation emergencies will rest with the radiological health section of the Virginia Health Department. Offsite evacuation will be supervised by the State police upon recommendation by the Health Department. In the event that it is necessary to evacuate people working or living on the Hog Island Water Refuge at the tip of the Gravel Neck Peninsula, the U.S. Coast Guard will be notified. The Coast Guard may utilize its own boats or call for the Virginia Marine Resources Commission to supply transportation. As an alternate, the applicant has made arrangements which will permit air evacuation by helicopters from Fort Lee. If necessary, boat and ship traffic on the James River will be restricted from the area surrounding the station within a 3-mile radius by the U.S. Coast Guard. They will post vessels

near the western entrance to Goose Hill channel and near the northern entrance to the Rock Landing Shoal Channel to control traffic. The Virginia Commission of Game and Inland Fisheries will provide boats to patrol the area boundary, if required.

We have reviewed the description of the emergency plans contained in the Final Safety Analysis Report, as amended, and have concluded that it meets the requirements of Appendix E to 10 CFR Part 50 and is acceptable.

3.6 Industrial Security

The plant site will be protected by a security fence. Closed circuit television and intrusion alarms will be used to detect unauthorized individuals on the site. Access of persons to the normal main entrance of the administration building will be controlled by personnel stationed in the control room, or the receptionist. Several means of communication are availble between the station and offsite law enforcement agencies. We have reviewed the industrial security measures to be provided to protect against industrial sabotage at the Surry Power Station and have found that the applicant intends to meet the regulatory requirements stated in AEC Safety Guide No. 17, "Protection Against Industrial Sabotage." We have determined that acceptable industrial security provisions will be established for the Surry Power Station.

3.7 Technical Specifications

The Technical Specifications in a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the AEC. We have reviewed the proposed Technical Specifications in detail and have held numerous meetings with the applicant to discuss their contents. Modifications to the proposed Technical Specifications submitted by the applicant were made to describe more clearly the allowed conditions for plant operation. The finally approved Technical Specifications will be made part of the proposed operating license. Included are sections covering safety limits and limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls. On the basis of our review, we conclude that normal plant operation within the limits of the Technical Specifications will not result in potential offsite exposures in excess of the 10 CFR Part 20 limits. Furthermore, the limiting conditions for operation and surveillance requirements will assure that necessary engineered safety features will be available in the event of malfunctions within the plant.

4.0 QUALIFICATIONS OF APPLICANT

4.1 Organization and Technical Qualifications

The Surry station manager will be responsible for the safe operation of the Surry Power Station. The superintendent-station operations, supervisor-engineering services and the chemistry and health physics supervisor will report directly to the station manager. The operating supervisor, the supervisor-electrical maintenance, and the supervisor-mechanical maintenance will report to the superintendent-station operations. The operating shifts report to the operating supervisor. For one unit operation, each operating shift will include a senior licensed shift supervisor, a licensed control room operator, a licensed assistant control room operator, and an unlicensed auxiliary operator. The applicant originally proposed that the shift complement for two unit operation consist of a senior licensed shift supervisor, two licensed control room operators, two licensed assistant control room operators, and two unlicensed auxiliary operators. As a result of our review, the applicant has agreed to provide two senior licensed operators per shift in this complement. Three graduate engineers, the instrument supervisor and the instrument technicians report to the supervisor-engineering services. We have evaluated the adequacy of the plant staff and have concluded that it is satisfactory for operation of the facility.

All key supervisory personnel assigned for initial operation of the plant meet or exceed the minimum qualifications stated in the American National Standards Institute, N-18.1-1971 Standard, and all future supervisory personnel will be required to meet the standard by the Technical Specifications. In addition, the Technical Specifications require that either the station manager or the superintendent-station operations obtain a senior reactor operator license within 18 months after initial criticality. On this basis, we have concluded that the key supervisory personnel assigned to the plant meet the ANSI N18.1-1971 Standard and conform with the regulatory requirements of AEC Safety Guide No. 8 and are satisfactory.

The station manager reports to the VEPCO superintendent-production operations who in turn reports to the director, production operations and maintenance. Approximately 10 graduate level engineers working in production operations under the general superintendent of production and approximately 20 engineers employed for design review matters under the director-power station design are available for technical assistance to the station manager. We have concluded that this will provide adequate technical support for the Surry Power Station.

4.2 Review and Audit of Operations

The Station Nuclear Safety and Operating Committee, comprised of the station's supervisory staff, will have the responsibility for periodically reviewing all proposed procedures, results of tests and experiments, proposed changes to the Technical Specificiations, and all design changes. They are required to periodically review all operations to detect any potential safety hazards. Further, they will investigate all reported instances of departure from the Technical Specification limits and will make tentative determinations as to whether or not proposals involve unreviewed safety questions. They will review all abnormal occurrence reports, and perform special reviews and investigations as requested by the chairman of the System Safety and Operating Committee, discussed below. In addition, they will initiate periodic drills to test the effectiveness of the emergency procedures. They will meet as called by the station manager, but not less than monthly. In the event of a disagreement between the recommendations of the Station Nuclear Safety and Operating Committee and the actions comtemplated by the station manager, the course determined by the station manager will be followed with immediate notification to the superintendent-production operations and the chairman of the System Safety and Operating Committee.

The second review body, the System Safety and Operating Committee, composed of members from outside the plant staff, will review all proposed changes to the operating license, including the Technical Specifications, and the minutes of the meetings of the Station Nuclear Safety and Operating Committee to determine if matters considered by that committee involved unreviewed or unresolved safety questions. In addition, they will review proposed changes or modifications to systems or equipment that have a safety significance. They will also conduct periodic reviews of station operations, review all reported instances of departure from Technical Specifications, and perform special reviews and investigations as requested by company management or as the committee deems necessary. Further, they will review abnormal performance of plant equipment and anomalies, unusual occurrences and incidents which are reportable under the provisions of 10 CFR Part 20 and Part 50. In addition, they will review proposed tests and experiments and results thereof, when applicable, and occurrences of safety violations.

The System Safety and Operating Committee will consist of the director-production operations and maintenance, superintendent-production operations, director-power station design, director-nuclear services, and supervisor-nuclear

design. The committee membership will also include the manager of each nuclear generating station operating on the VEPCO system, or its designee, and at least one qualified nonaffiliated technical consultant. Consultant members will have equal vote with the permanent members of the Committee. The Station Safety and Operating Committee will meet as required by the chairman of the committee, but not less than quarterly.

The responsibilities and authority of the two committees are stated in the Technical Specifications. We have concluded that the provisions for the review and audit of the plant operations are acceptable.

4.3 Quality Assurance Program

4.3.1 Quality Assurance During Construction

During construction, the Commission's regulatory staff conducted a review of the quality assurance program in effect at the Surry Power Station to assure that the basic elements necessary in all quality assurance programs were being provided by the applicant and its associated contractors. This review indicated that organizational changes were made to strengthen the overall quality assurance program since the inception of the project. These included the following:

1. Reorganization of the Stone and Webster Engineering Corporation's quality assurance structure to provide a separate independent quality assurance effort.

2. Assignment by the applicant of a full time resident engineer to provide surveillance and audit of the Stone and Webster quality control effort onsite.

3. Establishment by Vepco of a full-time supervisory quality assurance engineer to provide surveillance over all facets of the total quality control effort.

The quality control program for the Surry Power Station is described in detail in Section 15.4.6 of the Final Safety Analysis Report. The program as described therein meets all the requirements of Appendix B to 10 CFR Part 50, Quality Assurance Criteria. However, inspection by the Division of Compliance during plant construction indicated that deficiencies did exist which were not identified by the quality assurance program and that the quality assurance program, in operation, did not fully comply with the description presented in the application. These were discussed with the applicant and corrective action was taken to remedy the deficiency. These actions included the following:

1. Improvement of radiograph identification and storage methods,

2. Improved radiographic inspection features,

3. Revision of welding procedures including changes in format,

4. Increased quality control manpower and equipment,

5. Improved weld control procedures,

6. Improved documentation methods for welding and nondestructive test results,

7. Improved welding electrode control procedures,

8. Development of positive programs to determine the acceptability of suspect welds,

9. The development and use by the applicant of detailed audit plans to review contractor performance, and

10. The hiring of outside consultants to provide technical expertise where needed.

In addition, the applicant has performed a detailed review of construction logs, correspondence and other job records to verify that no problem areas exist that have not been previously identified. We have concluded that because of the changes made for the quality assurance programs, the conformance of the programs to 10 CFR Part 50, Appendix B, and the improvements made in the implementation of the programs, there is reasonable assurance that the level of quality provided in the Surry plant is acceptable.

4.3.2 Quality Assurance Programs During Operation

Modifications to the station design that occur following the issuance of an operating license will be made only after review by the Station Nuclear Safety and Operating Committee, and, when required by the Technical Specifications, the System Safety and Operating Committee. The supervisor-quality assurance will assure compliance with approved design concepts and equipment specifications for reliability and safety. He will maintain complete surveillance over the design modification and monitor progress to insure compatibility with existing approved design standards. The applicant has stated that the quality assurance program during operation will meet the requirements of 10 CFR Part 50, Appendix B. The Division of Compliance will review the quality assurance procedures in

this regard and determine that the procedures, as developed, meet the requirements of 10 CFR Part 50, Appendix B prior to issuance of an operating license. Therefore, we conclude that the quality assurance program to be in effect during operation is acceptable.

5.0 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

The ACRS completed its review of the application for authorization to operate Surry Power Station Units 1 and 2 at its 140th meeting, December 9-11, 1971. A copy of the ACRS letter dated December 17, 1971, is attached as Appendix B. The letter contains several recommendations and observations. These matters have been considered in our evaluation in the sections of the report indicated. These include the following:

1. The use of a catalytic hydrogen recombiner to control the buildup of hydrogen in the containment that could follow in the unlikely event of a loss-of-coolant accident (Section 3.2.2.5),

2. Details of the programs and procedures implemented to aid in assuring maintenance of integrity of sensitized stainless steel piping (Section 3.4),

3. The adequacy of instrumentation for determining the course of postulated accidents (Section 3.5), and

4. Common mode failure and anticipated transients without reactor scram (Section 3.2.4.3).

The ACRS concluded in its letter that if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that the Surry Nuclear Generating Units 1 and 2 can be operated at power levels up to 2441 MW(t) without undue risk to the health and safety of the public.

6.0 FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to financial data and information required to establish financial qualifications for an applicant for a facility operating license are 10 CFR 50 Part 33(f) and 10 CFR 50, Appendix C. We have reviewed the financial information presented in the application and have concluded that the Virginia Electric and Power Company is financially qualified to operate the Surry Power Station, Units 1 and 2. A detailed discussion of the basis for our conclusion is presented in Appendix J.

7.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended

(Section 170 and related sections), the Commission has issued regulations in 10 CFR Part 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licensees or facilities such as power reactors under 10 CFR Part 50.

The Commission's regulations in Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also the holder of a license under 10 CFR Part 70 authorizing the ownership and possession for storage only of special nuclear material at a reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR Part 50), shall, during the interim storage period prior to licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement with the Commission. Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective date of the 10 CFR Part 70 license. Payment of an annual indemnity fee is required. Virginia Electric and Power Company has furnished to the Commission proof of financial protection in the amount of $1,000,000 in the form of a Nuclear Energy Liability Insurance Association policy (Nuclear Energy Liability Policy, facility form) No. NF-186.

Further, Virginia Electric and Power Company executed Indemnity Agreement B-45 with the Commission as of August 11, 1970, the effective date of its preoperational fuel storage licenses, SNM-1191 and SNM-1264. The applicant has paid the annual indemnity fee applicable to preoperational fuel storage.

Under the Commission's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished, and an indemnity agreement covering such operation (as distinguished from preoperational fuel storage only) has been executed. The amount of financial protection which must be maintained for the Surry Power Station, Units 1 and 2 (which has a rated capacity of more than 100,000 electrical kilowatts) is the maximum amount available from private sources, i.e., the combined capacity of the two nuclear liability insurance pools, which amount is currently $82 million. (This will increase to $95 million on March 1, 1972.) Accordingly, no license authorizing operation of the Surry Power Station, Units 1 and 2 will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, several days in advance of anticipated issuance of the operating license document, evidence in writing, on behalf of the applicant, that the present coverage has been appropriately amended so that the policy limits have been increased, to meet the requirements of the Commission's regulations for reactor operations.

Similarly, no operating license will be issued until an appropriate amendment to the present indemnity agreement has been executed. Virginia Electric and Power Company will be required to pay an annual fee for operating license indemnity as provided in our regulations, at the rate of $30 for each 1,000 kilowatts of thermal capacity authorized in its operating license. On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR Part 140 have been satisfied and that, prior to issuance of the operating license, the applicant will be required to comply with the provisions of 10 CFR Part 140 applicable to operating licenses, including those as to proof of financial protection in the requisite amount and as to execution of an appropriate indemnity agreement with the Commission.

8.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted will be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are United States citizens. The applicant is not owned, dominated, or controlled by an alien, a foreign corporation or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the requirements of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material for military purposes is involved. For these reasons and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

9.0 CONCLUSIONS

Based on our evaluation of the application as set forth above, we have concluded that:

1. The application for facility licenses filed by the Virginia Electric and Power Company, dated March 20, 1967, as amended (Amendments Nos 1 thru 32) complies

with the requirements of the Atomic Energy Act of 1954, as amended, (Act), and the Commission's regulations set forth in 10 CFR Chapter 1; and

2. Construction of the Surry Power Station, Units 1 and 2 (the facility) has proceeded and there is reasonable assurance that it will be completed, in conformity with Provisional Construction Permits Nos. CPPR-43 and CPPR-44, the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

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

4. There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter 1; and

5. The applicant is technically and financially qualified to engage in the activities authorized by this operation license, in accordance with the regulations of the Commission set forth in 10 CFR Chapter 1; and

6. The issuance of this license will not be inimical to the common defense and security or the health and safety of the public.

Before operating licenses will be issued to the Virginia Electric and Power Company for operation of either Unit 1 or Unit 2, the respective unit must be completed in conformity with the provisional construction permits, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power level must be verified by the Commission's Division of Compliance prior to license issuance. Further, before an operating license is issued, the applicant will be required to satisfy the applicable provisions of 10 CFR Part 140.

Domenic B. Vassallo, Chief
PWR Branch No. 1
Division of Reactor Licensing

APPENDIX A

CHRONOLOGY

REGULATORY REVIEW OF THE VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION UNITS 1 AND 2

1.	January 21, 1970	FSAR Received, (application for Operating License not received).
2.	February 9, 1970	Submittal of Part A, "Application for Provision Operating License," Amendment No. 13.
3.	February 11, 1970	Preliminary meeting to discuss philosophy of review of Surry FSAR.
4.	February 22, 1970	Submittal of Amendment No. 14, page changes for the FSAR, technical specifications, and Appendix B, Seismic design for the nuclear steam supply system.
5.	May 20, 1970	Meeting with applicant to discuss use of court reporters at Surry meetings.
6.	May 25, 1970	Submittal of Amendment No. 15, page changes for insertion into the FSAR.
7.	June 24, 1970	Meeting with applicant to discuss Surry FSAR.
8.	September 29, 1970	Request to applicant for additional information.
9.	October 1, 1970	Visit to site.
10.	December 1, 1970	Submittal of Amendment No. 16, Supplement, Volumes 1 and 2, Reports, Lateral Load Pile Test dated July 1968 and Preoperational Environmental Radiation Surveillance Program.

11.	December 22, 1970	Meeting with applicant to discuss instrumentation drawing review.
12.	January 14, 1971	Submittal of Amendment No. 17, additional and revised pages for FSAR.
13.	January 20, 1971	Meeting with applicant to discuss various aspects of plant design and operation.
14.	February 13, 1971	Submittal of Amendment No. 18, additional and revised pages for FSAR. Submittal of Amendment 18, to Part A "Application for Licenses" General Information portion.
15.	March 2, 1971	Submittal of Amendment No. 19, additional and revised pages for FSAR.
16.	March 18, 1971	Meeting with applicant to discuss technical specifications.
17.	April 1, 1971	Submittal of Amendment No. 20, partial answers to the AEC regulatory staff's oral requests for additional information of 12/22/70, 1/15/71, and 1/20/71.
18.	April 6, 1971	Site visit and continuation of electrical system review.
19.	April 14, 1971	Meeting with applicant to discuss technical specifications and pre-operational testing.
20.	April 20, 1971	Meeting with applicant to discuss stainless steel welding.
21.	May 3, 1971	Submittal of Amendment No. 21, revised and additional pages for the FSAR and answers to questions 4.17, 14.4, and 14.20 in response to the AEC regulatory staff's request for additional information of 9/29/70 and revised technical specifications in their entirety.

22.	May 18, 1971	Meeting with applicant to discuss meteorology.
23.	May 25, 1971	Meeting with applicant to discuss Surry weld reworking.
24.	May 28, 1971	Submittal of Amendment No. 22, revised and additional pages and deletion of pages for the technical specifications and FSAR.
25.	July 2, 1971	Submittal of Amendment No. 23, additional pages for FSAR and three reports.
26.	July 13, 1971	Meeting with applicant to discuss sensitized stainless steel at the Surry Plant.
27.	July 20, 1971	Meeting with applicant to discuss the qualifications of onsite operating staff.
28.	August 6, 1971	Submittal of Amendment No. 24, Part A, revised pages for the License Application and Part B, revised pages for FSAR.
29.	August 11, 1971	Meeting with applicant to discuss technical specifications, plant design, and emergency plans.
30.	August 25, 1971	Meeting with applicant to discuss technical specifications.
31.	September 30, 1971	Submittal of Amendment No. 25, revised pages for FSAR and reports on seismic design, spray pump motor, and ECCS.
32.	October 22, 1971	Meeting with applicant to discuss plant design and operation.
33.	October 22, 1971	Submittal of Amendment No. 26, revised pages for FSAR and report on sensitized stainless steel.

34.	November 3, 1971	Submittal of Amendment No. 27, revised pages for FSAR.
35.	November 4, 1971	Meeting with applicant to discuss plant design, operating staff, equipment, and environmental tests.
36.	November 11, 1971	Submittal of Amendment No. 28, additional information on plant design modifications, environmental testing, and other miscellaneous items.
37.	November 26, 1971	Request to applicant for additional information on sensitized stainless steel and requirements for reactor operators.
38.	December 7, 1971	Submittal of Amendment No. 29, Supplement B to Report SW-MER-1A, Sensitized Stainless Steel in Pressurized Water Reactor Applications, Surry Power Station.
39.	December 29, 1971	Submittal of Amendment No. 30, revised pages for FSAR, Report No. SW-MER-2A, "The Effect of Fissuring on Welds in Elevated Temperature Service," and "Preoperational Environmental Radiation Surveillance Program."
40.	January 19, 1972	Meeting with applicant to discuss Technical Specifications and Industrial Security Plan.
41.	February 1, 1972	Meeting with applicant to discuss spray pump motors.
42.	February 1, 1972	Submittal of Amendment No. 31, revised pages for FSAR.
43.	February 11, 1972	Submittal of Amendment No. 32, revised pages for FSAR.

APPENDIX B

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

December 17, 1971

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON SURRY POWER STATION UNITS 1 AND 2

Dear Dr. Schlesinger:

At its 140th meeting, December 9-11, 1971, the Advisory Committee on Reactor Safeguards completed its review of the application of Virginia Electric and Power Company for authorization to operate Surry Generating Units 1 and 2 at power levels up to 2441 MW(t). Unit 2 is expected to be ready for operation about six months after Unit 1. This project was considered at the 139th Committee meeting, November 11-13, 1971, and at Subcommittee meetings at the site July 1, 1971, and in Washington, D. C. on November 5, 1971. During its review, the Committee had the benefit of discussion with representatives of Virginia Electric and Power Company, Westinghouse Electric Corporation, Stone and Webster Engineering Corporation, the Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed. The Committee reported on the construction of these units in its letter of April 29, 1968.

Surry Units 1 and 2 are located in Surry County, Virginia on the Gravel Neck Peninsula which extends northwest from the south bank of the James River. The nearest boundary of Newport News, Virginia, population about 140,000, is approximately 4.7 miles southeast of the site. The high population density portion of Newport News is 17 miles from the plant. There is a large transient population in the area of the plant during summer months.

Each nuclear unit employs a pressurized water reactor in a three-loop nuclear steam supply system of essentially the same design as the H. B. Robinson Unit No. 2 and Turkey Point Units 3 and 4, previously reviewed.

The applicant states that he intends to operate Units 1 and 2 in such a manner as to assure that maximum fuel rod linear power density does not exceed 16.4 kw/ft at 10[illegible]% of the rated power of 2441 MW(t). Performance of the emergenc[illegible] core cooling system (ECCS) during postulated loss-of-coolant accidents has been re-evaluated in the light of results from the Commission's FLECHT program, experiments and analyses by the applicant and his contractors, and information developed by the Regulatory Staff in recent studies of ECCS. The Committee believes that the applicant's proposed manner of operation is acceptable.

The applicant proposes to use a catalytic hydrogen recombiner to control the buildup of hydrogen in the containment that could follow in the unlikely event of a loss-of-coolant accident. The recombiner will be operable by the end of 1972. In the meantime, a backup purging system, to be operable before plant startup, will be relied upon for hydrogen concentration control. The Regulatory Staff should assure itself that the criteria for these systems are consistent with those for other engineered safety features.

The applicant has reported that the stress relieving of much of the cold bent type 316 stainless steel piping, some of which is utilized in the emergency core cooling system, resulted in its becoming sensitized to potential stress corrosion cracking under certain conditions. The applicant proposes that an augmented in-service inspection program, together with certain additional, special operating procedures, be implemented to aid in assuring maintenance of integrity of this piping throughout plant life. The details of these programs and procedures should be resolved in a manner satisfactory to the Regulatory Staff.

The applicant should assure himself that instrumentation for determining the course of postulated accidents is on hand at the station and that appropriate calibration methods and calculated bases for interpreting instrument responses are available.

The Committee reiterates its previous comments concerning the need to study further means of preventing common mode failures from negating reactor scram action, and design features to make tolerable the consequences of failure to scram during anticipated transients. The Committee believes it desirable to expedite these studies and to implement in timely fashion such design modifications as are found to improve significantly the safety of the plant in this regard. The Committee wishes to be kept informed of the resolution of this matter.

Other problems relating to large water reactors, which have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports, should be dealt with appropriately by the Regulatory Staff and the applicant as suitable approaches are developed.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that the Surry Nuclear Generating Units 1 and 2 can be operated at power levels up to 2441 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

Spencer H. Bush
Chairman

References

1. Amendment No. 12, dated January 21, 1970: Final Safety Analysis Report, Surry Power Station Unit Nos. 1 and 2, Volumes 1 through 4 and Supplements Volumes 1 and 2

2. Amendment No. 13, dated February 9, 1970: Amended License Application for Surry Power Station Unit Nos. 1 and 2

3. Amendments Nos. 14 through 29, to License Application for Surry Power Station Unit Nos. 1 and 2

4. "Security Program, Surry Power Station," (Proprietary), dated October 22, 1971, Virginia Electric and Power Company

APPENDIX C

Comments on

Surry Power Station Units 1 and 2
Virginia Electric and Power Company
Final Safety Analysis Report
Amendment No. 12 dated 12/1/69
Amendment No. 14 dated 2/27/70
Amendment No. 16 dated 12/1/70
Amendment No. 23 dated 7/2/71.

Prepared by

Air Resources Environmental Laboratory
National Oceanic and Atmospheric Administration
August 9, 1971

The important assumption in analyzing the meteorological aspects of the design basis accident is that all containment leakage occurs at ground level over a period of about 40 minutes. As a consequence, we have used the 20-ft Hog Island wind data rather than the 150-ft Surry Tower wind data in our evaluation.

Near ground level (20 feet) the average annual wind speed at the site is 6 mph with a 3.5 percent frequency of 'calms'. We have assumed, as did the applicant, that a wind speed listed as calm is any speed less than 1 mph. According to the tabular data in Amendment 23, page A-12 of NUS-TM-S-125, the frequency of calms at night is 3.0 percent of all observations.

Although we agree that the use of horizontal wind fluctuation data (sigma theta) is an appropriate technique to quantify atmospheric diffusion rates under cetain conditions, it is not appropriate under "non-steady" conditions as identified by Slade /1/. We have serious reservations that the variance computer discussed by the applicant can identify these non-steady conditions and that, in fact, the measurements of sigma theta under these conditions are meaningful (see Slade /1/, page 10, bottom paragraph). The sigma theta approach, in order to be meaningful and measurable by a variance computer, requires a statistically normal distribution of wind direction over the time period of interest. Furthermore, as Slade /1/ also points out, while non-steady wind indicates a large effective value of lateral spreading, the limited vertical dispersion at night with the possibility of local pooling of the effluent suggests the existence of regions of locally high concentrations. We feel this is particularly important in the case of short-term (40 minutes, for example) releases at distances of less than a mile from the source.

To evaluate the short-term (0-2 hr. release) consequences of the design basis accident, we have used only the wind speed distribution data from the Hog Island measurement location. We identified the 3 percent occurrence of calm conditions at night to be equivalent to a diffusion rate of Pasquill Type F and a wind speed of 0.5 m/sec. Extrapolating this worst condition to one which would occur at the 5 percent frequency level, we have used a wind speed of 1 m/sec and Type F. Assuming further a building wake factor of $c = 1/2$ and $A = 1290\ m^2$ the concentration at the exclusion distance of 500 m was calculated to be 9×10^{-4} sec m^{-3}. This value is in close agreement with that shown in Case A of Table IX of NUS-TM-S-125. To evaluate the mean annual concentration, we again used the Hog Island data as depicted by the annual wind rose on page A-12. We assumed the greatest annual frequency of a wind direction in any 22-1/2 degree sector was 10 percent, which included a prorating of the frequency of calms into the 16 sectors. Thus for a ground release, a mean diffusion rate equivalent to Pasquill Type D (neutral stability), and a 3 m/sec mean annual wind speed, the mean annual concentration at the nearest site boundary was calculated to be 7×10^{-6} sec m^{-3} also using the 20-ft wind data.

In summary, we have used the Hog Island wind speed distribution (including calms) by night and day and the general wind direction frequency to calculate the site boundary concentration which would be exceeded 5 percent of the time and the annual concentration for a routine release. Both values are in close agreement with the applicant's values as stated in Amendment 23, although the method of calculation differed.

Reference

/1/ Slade, D. H., 1966, Estimation of Dispersion from Pollutant Releases of a Few Seconds to 8 Hours in Duration. ESSA Technical Note 39-ARL-3, 26 pp.

APPENDIX D

REPORT ON THE SITE SEISMICITY FOR THE SURRY POWER STATION, VIRGINIA

At the request of the Division of Reactor Licensing of the Atomic Energy Commission, the Seismology Division of the Coast and Geodetic Survey has evaluated the seismicity of the area around the proposed reactor site in Surry County, Virginia, and has reviewed a similar analysis made by the applicant in the "Preliminary Safety Analysis Report," of the Virginia Electric and Power Company. The applicant's report of the seismicity which necessarily included an exhaustive valuation of the local geology, the surface foundation conditions, and the possibility of liquefaction is satisfactory for an evaluation of the seismic factor.

In reviewing the seismicity two significant factors were considered, first, the differences in maximum intensities in the Piedmont Province and the Coastal Plain Province in which the proposed reactor site is located, and, second, the intensities of distant earthquakes, New Madrid, Missouri, of 1812 and Charleston, South Carolina, of 1886. None of the Piedmont earthquakes in Virginia of intensity VII (MM) at distances of 90 to 250 miles from the proposed site were felt in the vicinity of the site. Within 50 miles of the

site in the Coastal Plain Province the reported earthquake intensities do not exceed IV and approximately 25 earthquakes of intensities I-IV have been reported since 1925.

In examining the effects of the New Madrid and Charleston earthquakes at the proposed site intensities ranging from V to a weak VI were reported at Williamsburg, Norfolk and Jamestown. A well documented intensity VI was reported at Richmond, approximately 50 miles from the proposed Surry site.

The Coast and Geodetic Survey, after considering the site geology and the proposed soil conditions, agree with the applicant that an acceleration of 0.07 g at ground surface would be adequate for representing earthquake disturbances likely to occur within the lifetime of the facility. In addition, we agree that the applicant's proposal to design for an acceleration of 0.15 g at ground surface would adequately protect the facility for ground motion from the maximum earthquake likely to affect this site.

U. S. Coast and Geodetic Survey
Rockville, Maryland 20852

February 23, 1968

APPENDIX E

Surry Power Station
Surry County, Virginia
AEC Docket 50-280, 281

Hydrology

The site is located on a peninsula formed by 180° turn of the James River between Jamestown and Newport News and about 25 miles upstream from Chesapeake Bay. The river is brackish in this reach and flow reverses with the tide.

Data of freshwater runoff past the site as shown in the Preliminary Safety Analysis report appear to be reasonable. The data however do not include the drought of 1930-31 during which flows occurred that were considerably lower than the minimum monthly flow of 857 cfs shown. Thus the mean flow for the period August through October 1930 was less than 700 cfs (cubic feet per second), lowest mean monthly flow was about 620 cfs and minimum flow for 7 consecutive days was about 440 cfs on the basis of records of James River at Cartersville and Appomattox River near Petersburg.

Except during periods of high runoff, tidal flow primarily accounts for the upstream and downstream currents past the site. The oscillatory flow does not directly indicate the rate of flushing out from the James Estuary of effluents added to it. In general, flushing out of effluents that remain near the surface would be at a greater rate than is indicated by the runoff, while effluents that sink to a lower depth of the river would be flushed out at a slower rate. No quantitative data are presented in the report but it may be estimated that heated effluent from the plant would remain near the surface and would be flushed out at a rate that is considerably greater than the runoff would indicate.

The site is underlain by unconsolidated beds that dip toward the sea. Ground-water is utilized from several deeper sandy strata in these beds as well as from the uppermost quaternary deposits. The latter are extensively tapped for domestic or other small volume uses, except near the James River where pumping would lead to the intrusion of brackish water. In the lower strata water is found at artesian pressure because of overlying less permeable deposits. Water enters these strata principally near the Fall line where they are exposed. Wells along the James River between Claremont and Norfolk have been drawing water from this artesian aquifer since the early part of this century and the artesian head has been dropping steadily. Where this head is lower than the overlying surficial ground-water table local recharge from the upper ground water through the confining deposits to the artesian aquifer can occur.

Radionuclides spilled at the site or deposited on the ground in the vicinity would be expected to enter the surficial ground water and migrate slowly towards the nearest stream channel. Entry into the lower aquifer could occur in areas where pumping has lowered the artesian head. Migration times through the less permeable confining beds is a matter of conjecture but could be expected to be a matter of decades.

The nearest well to the site is at Hog Island State Water Fowl Refuge about one mile from the reactor. Other nearby wells are at Jamestown, Fort Eustis, and Bacons Castle.

References:

Cederstrom, D. J., 1945, Geology and Ground-Water Resources of the Coastal Plain of Southeastern Virginia: Virginia Geol. Survey, Bull. 63.

_____, 1957, Geology and Ground-Water Resources of the York-James Peninsula, Virginia: U. S. Geol. Survey, Water Supply Paper 1361.

Geology

The analysis of the geology of the Surry Nuclear Power Station site, Atomic Energy Commission Docket (50-280, 281) was reviewed and compared with the available literature. The geologic analysis is carefully derived and presents an adequate appraisal of those aspects of the geology which would be pertinent to an engineering evaluation of the site.

Although there are no identifiable geologic structures which could be expected to localize seismicity in the immediate vicinity of the site, the area is underlain at depth ca ~ 1500 feet by crystalline rocks continuous with those which crop out to the west within the Piedmont Province. Hence it must be assumed that earthquakes with bedrock intensities comparable to those characteristic of the Piedmont Province may occur in the area of the plant.

APPENDIX F

DEPARTMENT OF THE ARMY
COASTAL ENGINEERING RESEARCH CENTER
5201 LITTLE FALLS ROAD, N.W.
WASHINGTON, D.C. 20016

CEREN

9 November 1971

Mr. Richard C. De Young
Asst. Director for Pressurized Water Reactors
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. De Young:

Reference is made to your letters regarding Docket Nos. 50-280 and 50-281 Virginia Electric and Power Company's proposed Surry Power Station Units 1 and 2.

Pursuant with our arrangements Mr. R. A. Jachowski of the CERC staff has reviewed all pertinent information contained in the Final Safety Analysis Report (FSAR) and subsequent amendments thereto from the standpoint of establishment of the design flood protection level. This included the review of the storm surge associated with the Probable Maximum Hurricane (PMH) and wind wave and wave runup analysis.

We concur with the applicant's computed value of Elevation 22.3 feet above MSL for the maximum probable still water level at the station site. We also concur in the Elevation of 28.2 feet above MSL for the limit of significant wave runup on the 1 on 5 multi-layer rock reveted embankment. We further concur in the elevation of 36.2 feet above MSL determined for significant wave runup against the emergency pumphouse.

We further concur that the depressed river level resulting Probable Maximum Hurricane poses no operational problems as the high level intake canal contains sufficient storage to maintain the station in a safe shut down condition. If you have any further questions regarding this matter please let us know.

Sincerely yours,

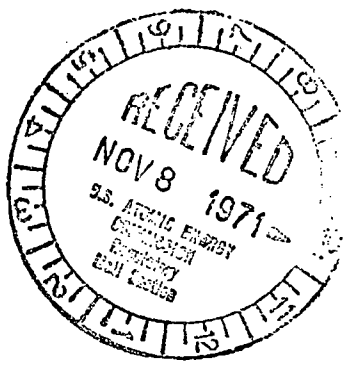
DON S. McCOY
Lieutenant Colonel, CE
Director

APPENDIX G

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

4 November 1971



Mr. Edson G. Case, Director
Division of Reactor Standards
U.S. Atomic Energy Commission
Washington, D.C. 20545

Re: Contract No. AT(49-5)-2667
Final Report
Surry Power Station Units 1 and 2
Virginia Electric and Power Company
AEC Docket Nos. 50-280 and 50-281

Dear Mr. Case:

We are transmitting herewith 8 signed copies of our Final Report concerning the Operating License Review for the Surry Power Station Units 1 and 2.

Sincerely yours,

W. J. Hall

pg
Enclosure

cc: N. M. Newmark

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

REPORT TO THE AEC REGULATORY STAFF

STRUCTURAL ADEQUACY

OF

SURRY POWER STATION UNITS 1 AND 2

Virginia Electric and Power Company

AEC Docket Nos. 50-280 and 50-281

by

N. M. Newmark

and

W. J. Hall

Urbana, Illinois

4 November 1971

REPORT TO THE AEC REGULATORY STAFF

STRUCTURAL ADEQUACY

OF

SURRY POWER STATION UNITS 1 AND 2

INTRODUCTION

This report is concerned with the structural adequacy of the containment structures, piping, equipment and other critical components for the Surry Power Station Units 1 and 2 for which application for a construction permit and an operating license has been made to the U.S. Atomic Energy Commission by the Virginia Electric and Power Company. The facility is located in Surry County, Virginia on the south side of the James River, approximately 30 miles NW of Norfolk, Virginia and 7 miles S of Williamsburg, Virginia.

This report is based on a review of the Final Safety Analysis Report and supplements (Ref. 1) and supplementary material (Ref. 2). The report also is based in part on discussions held with the applicant and his consultants at the time of a visit to the Surry Power Station Site on 20 October 1970 by W. J. Hall with D. Muller and R. Lee (DRL/AEC). A discussion of the adequacy of the structural criteria presented in the Preliminary Safety Analysis Report is contained in our report of March 1968 (Ref. 3), and unless otherwise noted no comment will be made in this report concerning points covered therein.

The design criteria for Class I components for this plant called for a design to withstand a Design Basis Earthquake of 0.15g maximum horizontal ground acceleration, coupled with other appropriate loadings to provide for containment and safe shutdown. The plant also was designed for an Operating Basis Earthquake of 0.07g maximum horizontal ground acceleration acting simultaneously with other appropriate loadings forming the basis of containment design.

COMMENTS ON ADEQUACY OF DESIGN

Foundations

The foundation conditions for the various Surry Power Station structures were considered in detail at the time of the PSAR review. At the time of the site visit it was ascertained that, with Unit 1 of the plant at approximately 80% completion, no unexpected motions of the foundations had been noted under the static loading imposed.

In the interim, since the PSAR review, two lateral load tests of piles were made to verify the computed deflections used in the design of the fuel building. The pile tests showed that the computed values were reasonable; the values of possible lateral motion used in design are presented in Table 2.4-5 for the piling. These values appear reasonable to us.

The calculated expected displacements for the various structures are presented in Table 2.4-5; the rattle space provided between buildings is presented on page 2.4.8-3, namely a space of 6 in. between the pile-supported fuel building and the auxiliary building and between the fuel building and the containment structures, and a clearance of 3 in. between other structures. The applicant indicates on page 15.2-17 that the clearance provided between structures is a minimum of 2 in. In any event, whether the clearance is 2 in. or the larger values as cited in Section 2 on page 2.4.8-3, we feel that these clearances are adequate.

The applicant advises in the answer to Question 2.14 that a check of the vertical motions has also been made, and that a further check has been made to insure that the pipe stresses of all Class I piping between structures is within acceptable limits. We concur in the approaches followed.

A discussion of the analyses that have been carried out for liquefaction potential of strata in the site area is presented in Section 2.4.5.1. The summary

of the liquefaction studies indicates that with the depressed piezometric levels under the Surry structures, the potential for liquefaction is quite low. Additional information on the depressed piezometric levels is presented in the answer to Question 2.16, wherein it is noted that the piezometric levels are slightly lower in 1971 in comparison to the values for 1967. These data, covering piezometer readings over the plant area, help confirm the low probability of any difficulty with liquefaction. Also, the applicant states that the liquefaction analysis was based on surface accelerations of 0.15g. On the basis of the information provided, and other studies and judgments that we have made independently, we believe the design has been carried out in a manner which makes the probability of liquefaction negligible. We believe the design to be satisfactory in this regard.

Dynamic Analyses

a) Class I Structures

The procedures followed in the dynamic analysis of major Class I structures are described in Section 15.5.1.4 of the FSAR and in the answer to Question 5.21. It is noted that the structures were analyzed using a Stone and Webster proprietary computer program, "Container Vessel Seismic Analysis". It is indicated on pages 15.5.1.5-2 and -3, in the answers to Questions 5.12 and S5.12, and Ref. 2(c), that for the OBE a value of 5 percent of critical damping was used for the coupled soil-structure model, whereas for the Design Basis Earthquake a damping value of 10 percent of critical was used; these latter values, according to the applicant, include the effects of radiation damping and soil-structure interaction. For example, the high damping for Mode 2, as presented in the answer to Question S5.12-1, corresponds only to translational motion and is not applicable to other modes or to internal strains. On the basis of the information presented in the answer to Question 5.12, and

the discussion in Section 15.5, and on the basis of additional clarifying material to be provided in Section 15.5 in a later amendment, we conclude that structural component damping of 5 percent was used for the DBE analysis. The total modal damping (by mode) tabulated in the answer to Question 5.12 appears reasonable for the Surry site. The system damping of 10 percent for the DBE, as discussed in Ref. 2(c), includes the energy losses from the coupled soil-structure model and appears reasonably conservative to us for this site.

b) Piping and Equipment

The seismic design approach for the nuclear steam supply system is outlined in Appendix B of the FSAR. It is indicated that the Westinghouse portions of the system are analyzed in accordance with Westinghouse Report WCAP-5890, Rev. 1, and we are in agreement with the approach adopted.

The applicant states in the answer to Question 5.12 that 0.5 percent damping was used in all the piping analyses for the OBE and 1.0 percent for the DBE; we have been advised that the 1.0 percent of critical value is correct, in contrast to the lower value stated on page 24 of Ref. 2(c). A general description of the analysis procedure employed by NSSS piping is presented in the answer to Question 4.12.

With respect to the dynamic piping analysis, the answer to Question 5.1 indicates that differential support motions for piping were included in the analyses. Where piping ran between buildings and the supports could move relative to one another, the differential motion also was taken into account in the analyses.

For piping and components supplied by Stone & Webster, the answer to Question 4.10 indicates that major equipment-supporting structures and piping systems were dynamically analyzed. Mechanical components (Class 1) were procured with specified seismic coefficients corresponding to the peak acceleration values

on the response spectrum curves. Additional information on these approaches, including typical stress summaries, is presented in Ref. 2(c). The procedures and results presented appear reasonable and satisfactory to us.

c) Vertical Earthquake Excitation and Critical Instrumentation and Controls

The answer to Question 4.23 describes the approach employed for the vertical excitation. For the containment structure and other rigid structures, the approach of using two-thirds of the horizontal acceleration is satisfactory. In the case of equipment, it is indicated that the acceleration magnitudes are included in the equipment specification. Subsequently later during plant design and construction, a seismic review was carried out as described in detail in Ref. 2(c). Floor response spectra were computed and adapted to design use. As noted on page 24 of Ref. 2(c), in Ref. 2(d) and in the answer to Question S5.16-1, many changes and alternations in equipment support and bracing were made. The margins reported in Table 4-1 are satisfactory.

The answer to Question 4.10 indicates that the vendor is required to validate component integrity under the specified seismic conditions. A general description of the approach adopted is presented in Sections 5.0 and 6.0 of Ref. 2(c) and is acceptable to us.

Reactor Internals

The answer to Question 4.13, dealing with reactor internals, makes reference to the H. B. Robinson response to Question 1.1B dated 5 November 1969. Topical Report WCAP #7332-L, covering the seismic analyses of reactor internals for pressurized water reactors, is applicable. The approach followed here appears acceptable to us.

Design Stresses

The discussion on page 15.2-17 of the FSAR indicates that for those structures subjected to the DBE loading and analyses, allowable stresses do not

exceed 90 percent of the yield strength for structural steel or in the case of reinforcing steel and concrete, the capacity reduction factor times the specified yield strength for reinforcing steel or the specified strength for concrete. The answer to Question 5.8 indicates that the stresses cited are of the order of 50 percent of ultimate strength. On this basis there is reason to believe that the deformation associated with these design stresses are generally low although the amount of deformation that may be controlling under the maximum loading conditions between the various materials used may not be precisely similar. We must assume, for lack of other evidence, that where composite action is called for, the compatibility of the deformations in the various elements have been studied carefully and taken into account.

SUMMARY COMMENT

On the basis of our review of the FSAR and the answers to the questions given in the supplement, it appears to us that the design possesses a margin of safety which is adequate in terms of provision for safe shutdown for a Design Basis Earthquake of 0.15g maximum horizontal ground acceleration and to withstand otherwise the effects of an earthquake of half this amount.

REFERENCES

1. "Final Safety Analysis Report -- Vol. 1-5 (Part B), plus Supplement Vol. 1 and 2, and Amendments 14, 15, 17, 19, 20, 21, 22, 23, 24, 25", Surry Power Station Units 1 and 2, Virginia Electric and Power Company, AEC Docket Nos. 50-280 and 50-281, 1969.

2. Supplementary material:

   (a) "Surry Nuclear Power Station Preoperational Environmental Radiation Surveillance Program", Report, 1 May 1968 through 30 June 1970, Virginia Electric and Power Company.

   (b) "Lateral Load Pile Tests -- Surry Power Station -- Virginia Electric and Power Company", prepared by Stone & Webster Engineering Corporation, Boston, Massachusetts, July 1968.

(c) "Seismic Design Review, Equipment and Piping, Surry Power Station", Virginia Electric and Power Company, Revised 15 Sep. 1971.

(d) "Qualification Test of Prototype -- Vertical Induction Motor -- Containment Recirculation Spray Pump", by J. P. Waggener et al, Franklin Institute Research Laboratories, Final Report F-C2909, 1971.

3. "Adequacy of the Structural Criteria for the Surry Power Station Units 1 and 2, Virginia Electric and Power Company", AEC Docket Nos. 50-280 and 50-281, by N. M. Newmark, W. J. Hall, and A. J. Hendron, Jr., March 1968.

APPENDIX H

ADDRESS ONLY THE DIRECTOR,
BUREAU OF SPORT FISHERIES
AND WILDLIFE

United States Department of the Interior
FISH AND WILDLIFE SERVICE
BUREAU OF SPORT FISHERIES AND WILDLIFE
WASHINGTON, D.C. 20240

50-280
50-281

OCT 4 1971

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OCT 5 1971
U.S. ATOMIC ENERGY COMMISSION
Regulatory
Mail Section

Mr. Harold L. Price
Director of Regulation
U.S. Atomic Energy Commission
Washington, D.C. 20545

Dear Mr. Price:

This letter contains our Bureau's comments on the Virginia Electric and Power Company's application for an operating license for the Surry Power Station, Units 1 and 2, James River, Virginia, AEC Docket Nos. 50-280 and 50-281. The Final Safety Analysis Report and all subsequent amendments through No. 22 also were reviewed in conjunction with the application. The comments of the Mid-Atlantic Coastal Fisheries Research Center, NMFS, Department of Commerce, were obtained and are included in this review.

Former Acting Commissioner Puncochar's letter of comment on the construction permit, dated September 8, 1967, described the physical features of the project, discussed the fish and wildlife resources that would be affected, and recommended measures to protect these resources from significant damage. Former Commissioner Pautzke's letters of February 12 and April 9, 1968, provided additional comments. The construction permit was issued on June 25, 1968. We understand that construction of the project is substantially complete and fuel loading is scheduled for January 1972.

The preoperational radiological monitoring program initiated in 1968 is generally satisfactory. However, it should be expanded to include representative species of carnivorous and herbivorous fishes and aquatic vegetation collected within 500 feet of the discharge point. With these additions, the radiological monitoring program would be satisfactory insofar as the fish and wildlife resources are concerned. It will be necessary to conduct postoperational radiological monitoring surveys to determine the extent of radioactivity buildup, if any, in the aquatic and terrestrial organisms.

Operation of the project has the potential of affecting the aquatic resources of the James River estuary adversely. We are concerned that damage to aquatic life may result from the release of heated condenser cooling water and that a significant number of aquatic organisms may become entrained in the cooling water intake structure and destroyed.

Based on records for the period from October 1934 to September 1965, the applicant determined that the minimum mean-monthly fresh-water discharge for the James River at the project site was 857 cfs. During the period August through October 1930, an interpretation of upstream records indicates that mean flow was less than 700 cfs with the lowest mean monthly flow being about 600 cfs. During this drought, the flow was about 440 cfs for seven consecutive days.

As in most estuaries, the oscillatory tidal flow is much greater than the fresh-water flow. Being oscillatory, such flow flushes contaminants out of the estuary only by dispersion--since most of the same water moving seaward on the ebbtide returns upstream on the flood tide. Dispersive flushing may be a very slow process, especially where tidal range in stage is small, such as in the James River estuary.

Superimposed upon the oscillatory tidal flows and the fresh-water flows, there is a net nontidal circulation caused by intrusion of saline water within the estuary. The applicant states that this circulatory flow is several-fold larger than the fresh-water discharge. Of the tidal, fresh-water, and circulatory discharges, convective transport of contaminants by the last two may be the most important.

The applicant used a physical model to aid in designing the cooling water intake and discharge systems and to evaluate temperature effects in the estuary near the plant site. The design of the station discharge is intended to cause the water, which has been heated by 14° F, to mix and thereby become reduced rapidly in temperature. Based on these model tests, the applicant states that a maximum temperature rise of 1.4° F may be realized at the far shore under summer conditions.

A paper entitled "Interpretation of Water Pollution Data from Tidal Estuary Models" by D. R. F. Harleman, E. R. Holley Jr., and W. C. Huber, published in the Third International Conference on Water Pollution Research, Vol. 3, 1966, indicates that the results of dispersion studies, such as the movement of heated water and radionuclides, may be invalid when based on physical models.

The reliance on a physical model for evaluating temperature effects, the failure to consider the worst low fresh-water flow conditions, and the possible overestimation of the ability of tidal discharges to flush contaminants from the estuary appears to make the applicant's evaluation of the thermal effect questionable.

In view of the above, postoperational monitoring of water temperatures will be necessary to determine the impact of the heated effluent on aquatic resources. The company has expressed assurance that the problem referenced above will be resolved satisfactorily. Therefore, we

APPENDIX I

ASSUMPTIONS USED IN CALCULATIONS OF OFFSITE DOSES FROM DESIGN BASIS ACCIDENTS

A. Loss-of-Coolant Accident Assumptions

Except for the following, all assumptions are the same as AEC Safety Guide No. 4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

1. Power level of 2546 MWt.

2. Design containment leak rate of 0.1% per day for the first hour, no leakage thereafter.

3. Meteorological conditions consisting of Pasquill Type F stability and a 1 meter per second wind speed.

4. Building wake factor, C = 0.5, and the reactor building cross sectional area, A = 1870 square meters.

5. Dose reduction factor of 2.0 for the thyroid dose, by the sodium hydroxide containment spray.

6. Whole body dose includes beta plus gamma contributions.

B. Refueling Accident Assumptions

1. Reactor power level of 2546 MWt.

2. The accident occurs after 100 hours of decay.

3. One fuel assembly fails (204 fuel rods).

4. 10% of the noble gases released from the assembly.

5. 10% of the halogens released from the assembly.

6. 1% of the iodine released from the fuel gap is released from the fuel pool water.

7. The iodine above the pool is 75% elemental and 25% organic.

8. 90% of the elemental iodine and 70% of the organic iodide in the building air is removed by the spent fuel pool area charcoal filters.

9. Fuel assembly peaking factor of 1.65.

10. Breathing rate of 3.47×10^{-4} cubic meters per second.

11. Meteorology as in loss-of-coolant accident assumptions (Section A of this Appendix).

12. Semi-infinite cloud dose model.

13. Whole body dose includes beta plus gamma contributions.

C. Gas Decay Tank Rupture Accident Assumptions

1. Gas decay tank content is 94,500 curies equivalent of Xe-133 which the applicant has stated will be the maximum fission product inventory in any tank.

2. Semi-infinite cloud model used.

3. Meteorology as in loss-of-coolant accident assumptions with no credit for building wake effect. (Section A of this Appendix.)

4. Whole body dose includes beta plus gamma contributions.

D. Steam Line Break Accident Assumptions

1. Reactor operating with its design source term (1% failed fuel).

2. Primary to secondary steam generator total leakage of 10 gpm in the two unaffected steam generators.

3. Secondary coolant system halogen inventory at equilibrium and proposed Technical Specification limit set to yield a 1.5 rem thyroid dose at the exclusion radius due to a loss-of-load accident.

4. Steam line failure releases 170,000 pounds of the secondary coolant system volume to the atmosphere, and all halogens flash to the atmosphere.

5. Remainder of secondary coolant released to atmosphere during plant cooldown using remaining three steam generators to remove decay heat. The release is through the atmospheric relief valves since loss of offsite power is assumed.

6. Partition factor of 10 for halogens released from relief valves.

7. During 8 hour cooldown the primary coolant carried over to the secondary side is also released from the relief valves.

8. Meteorology and breathing rate as in loss-of-coolant accident assumptions (Section A of this Appendix).

9. Semi-infinite cloud model.

10. Whole body dose includes beta plus gamma contributions.

E. Steam Generator Tube Rupture Accident Assumptions

1. Reactor is operating at full power of 2546 MWt.

2. A steam generator tube fails, releasing 115,000 pounds of the reactor coolant to the secondary side during the 30 minute isolation time.

3. Offsite power is lost, and plant cooldown is achieved by relieving the secondary steam through the atmospheric relief valves;

4. Reactor operating with its design source term (1% failed fuel).

5. Secondary coolant system halogen inventory at equilibrium and proposed Technical Specification limit set to yield a 1.5 rem thyroid dose at the exclusion radius in the event of a loss-of-load accident.

6. All halogens from the primary and secondary coolant are released to the atmosphere with a partition factor of 10 through the relief valves.

7. Meteorology and breathing rate as in loss-of-coolant accident assumptions (Section A of this Appendix).

8. Semi-infinite cloud model.

9. Whole body dose includes beta plus gamma contributions.

APPENDIX J

FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to the financial data and information required to establish financial qualifications for an applicant for an operating license are 10 CFR 50.33(f) and 10 CFR 50 Appendix C. The application of Virginia Electric and Power Company (VEPCO), as amended, and the accompanying certified annual financial statements provide the financial information required by the Commission's regulations.

These submittals contain the average annual operating costs of the Surry Power Station, Units 1 and 2, plus the estimated costs of permanently shutting down the facility and maintaining it in a safe shutdown condition. The estimated annual levelized costs over the life of the facility (30 years) for operating Units 1 and 2 are $44.6 million and $33.9 million, respectively. These estimates include the following costs: carrying charges, taxes, operation, maintenance, insurance, and fuel. The applicant's estimate of the cost of permanently shutting down the facility (both units) as specified in Amendment 30 to the application is $1.5 million for the first year and $200,000 annually thereafter for maintaining each unit in a safe shutdown condition.

We have examined the certified financial statements of Virginia Electric and Power Company to determine whether it is financially qualified to meet these estimated costs. The information contained in

VEPCO's calendar year 1970 financial report indicates that operating revenues for 1970 total $374.9 million; operating expenses were $279.7 million, of which $46.8 million represented depreciation. The interest on long-term debt was earned 2.7 times; and the net income for the year was $72.1 million, of which $47.6 million was distributed as dividends to stockholders and the remainder of $24.5 million was retained for use in the business. As of December 31, 1970, the Company's assets totaled $1,829.5 million, most of which was invested in utility plants ($1,746.9 million); retained earnings amounted to $95.7 million. Financial ratios computed from the 1970 statements indicate a sound financial condition, e.g., long-term debt to total capitalization - .56, and to net utility plant - .53; net plant to capitalization - 1.05; the operating ratio - .75; and the rates of return on common - 11.2%, on stockholders' investment - 9.8%, and on total investment - 6.5%. The record of VEPCO's operations over the past 5 years reflects that operating revenues increased from $249.3 million in 1966 to $374.9 million in 1970; net income increased from $52.4 million to $72.1 million; and net investment in plant from $1,017 million to $1,747 million; while number of times interest earned on long-term debt declined from 3.9 to 2.7 times. Moody's Investors Service rates the Company's first mortgage bonds as Aa (high quality). The Company's current Dun and Bradstreet Credit Rating is 5A1 (the highest category).

Our evaluation of the financial data submitted by the applicant, summarized above, provides reasonable assurance that the applicant possesses or can obtain the necessary funds to meet the requirements of 10 CFR 50.33(f) with respect to the operation of the Surry Power Station, Units 1 and 2. A copy of the staff's financial analysis of the company is attached as Table J-1.

TABLE J-1

FINANCIAL ANALYSIS

	(dollars in millions) Calendar Year Ended December 31		
	1970	1969	1968
Long-term debt	$ 932.0	$ 762.0	$ 602.8
Utility plant (net)	1,746.9	1,456.6	1,243.3
Ratio - debt to fixed plant	.53	.52	.48
Utility plant (net)	1,746.9	1,456.6	1,243.3
Capitalization	1,668.0	1,365.1	1,147.8
Ratio of net plant to capitalization	1.05	1.07	1.08
Stockholders' equity	736.0	603.1	545.0
Total assets	1,829.5	1,531.2	1,301.2
Proprietary ratio	.40	.39	.42
Earnings available to common equity	64.4	57.7	53.6
Common equity	574.6	476.7	453.6
Rate of earnings on common equity	11.2%	12.1%	11.8%
Net income	72.1	63.3	57.8
Stockholders' equity	736.0	603.1	545.0
Rate of earnings on stockholders' equity	9.8%	10.5%	10.6%
Net income before interest	119.6	98.5	84.3
Liabilities and capital	1,829.5	1,531.2	1,301.2
Rate of earnings on total investment	6.5%	6.4%	6.5%
Net income before interest	119.6	98.5	84.3
Interest on long-term debt	44.1	33.6	24.9
No. of times long-term interest earned	2.7	2.9	3.4
Net income	72.1	63.3	57.8
Total revenues	399.3	340.6	304.2
Net income ratio	.18	.18	.19
Total utility operating expenses	279.7	242.1	219.9
Total utility operating revenues	374.9	326.4	297.5
Operating ratio	.75	.74	.74
Utility plant (gross)	2,082.5	1,754.8	1,508.6
Utility operating revenues	374.9	326.4	297.5
Ratio of plant investment to revenues	5.6	5.4	5.1

Capitalization:	1970 Amount	1970 % of Total	1969 Amount	1969 % of Total
Long-term debt	$ 932.0	55.9%	$ 762.0	55.8%
Preferred stock	161.4	9.7	126.4	9.3
Common stock & surplus	574.6	34.4	476.7	34.9
Total	$1,668.0	100.0%	$1,365.1	100.0%

Moody's Bond Ratings: First Mortgage Aa
Convertible Debentures A

Dun & Bradstreet Credit Rating: 5A1

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